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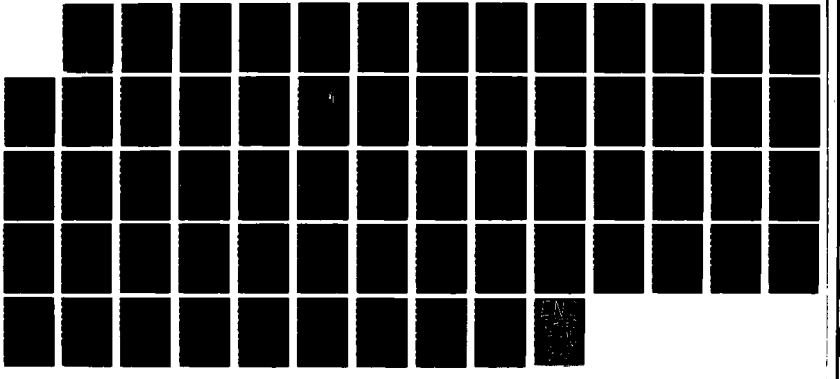

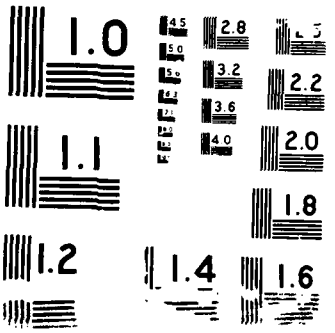
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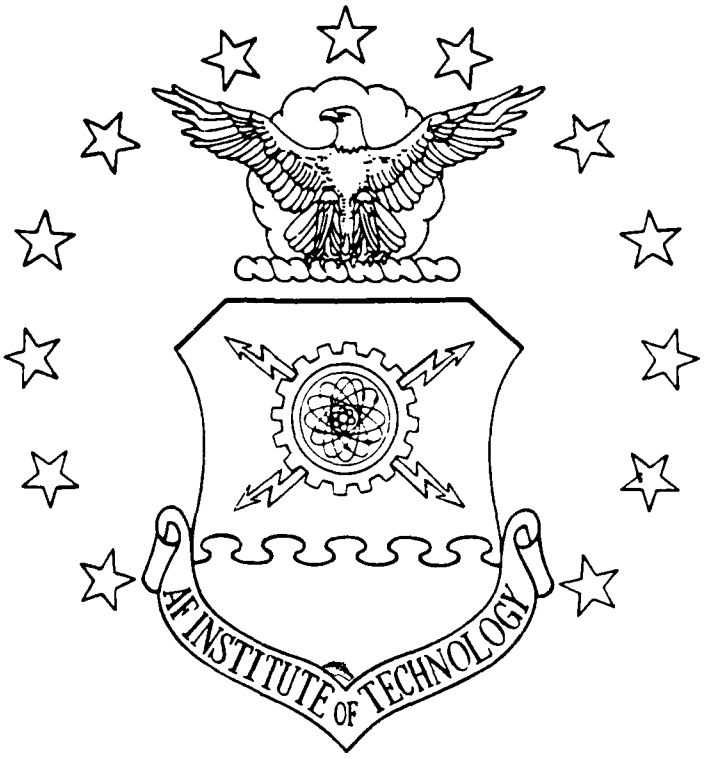

A METHOD OF DETERMINING NEUTRON DOSE TO A HUMAN PHANTOM

THESIS

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THESIS

Presented to the Faculty of the School of Engineering

of the Air Force Institute of Technology

Air University

In Partial Fulfillment of the

Requirements for the Degree of

Master of Science in Nuclear Engineering

Michael G. Archuleta, B.S.

Captain, USAF

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Preface

Often the research or military communities require a quick estimation of neutron radiation on human body organs. A method of estimation can help solve problems ranging from amount of radiation shielding required in a laboratory to distance required to protect air crews from their own nuclear bomb drops.

I would like to thank Major Beller for presenting a thesis topic I actually liked -- then letting me work on it. I also thank him for his help and guidance using the sometimes "trying" MORSE code. Special thanks to my cronies for putting up with me in times of stress and taking me out hunting and fishing for the much needed breaks. An added thanks to the mutts, J.D. and Magnum, for warming my feet while I typed and distracting me when I needed it. And above all, I thank my parents who made this all possible, and Vicki who made it all bearable.

- Michael G. Archuleta

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Abstract

This report describes a computer method of determining absorbed neutron dose to a human phantom. Modifications to the Oak Ridge National Laboratory MORSE Monte Carlo code result in a code capable of estimating absorbed dose on a human phantom in the standing position. The phantom organs analyzed are the skin, bone, brain, gastro-intestinal tract, and all remaining tissue. The organ choices are limited to organs capable of incapacitating a human. The code allows for five different source direction configurations that simulate neutrons, of any specified energy distribution, incident on the phantom.

MORSE analysis of a fission neutron spectrum on the phantom produces absorbed dose estimates comparable with Japanese atomic bomb survivor dose estimates by Scientific Applications International Corporation. The analysis of 24,000 source neutrons requires less than 15 central processing unit minutes on a VAX 11/780 computer (VMS operating system). Although the code is currently usable, additional phantom model orientations, energy-dependent quality factors, and implementation of secondary gamma-ray dose estimation could greatly improve the flexibility and usefulness.

A METHOD FOR DETERMINING NEUTRON DOSE TO A HUMAN PHANTOM

I. Introduction

Background

The nuclear age brought with it many new problems for both research scientists and military strategists alike. One such problem, and the focus of this research effort, is determining the absorbed radiation dose to a human from neutrons produced in reactors, weapons, or simple neutron sources. Effectively solving this problem for organs capable of incapacitation by neutrons could be of great interest, especially to military strategists, for the following and many more reasons:

1. A "safe" distance between nuclear targeted locations and ground troops must be known to ensure troop survivability in a theater of conflict.
2. Air crews delivering nuclear weapons to many targets must be able to reach minimum "safe" distances from detonation locations to survive effects from exposure to each successive detonation.
3. The amount of shielding required in above or below ground shelters must be known to protect individuals from other than "safe" radiation environments so that operations are sustained throughout combat conditions.

In all three cases, "safe" refers to the maximum amount of radiation the average human can withstand without suffering incapacitation during the length of a required duty (i.e. a B-52 bomber flight).

The scientific community possesses many methods of estimating neutron doses for specific cases such as the Hiroshima and Nagasaki atomic bomb victims

(6:610-"19). Other studies (11) provide dosimetry calculations from radiation effects correlations. The list of studies and estimations is endless. Unfortunately, all of the studies are specific and time consuming.

Problem and Scope

The problem is to determine a fast, convenient, and generic method of estimating the radiation dose to specific human organs from neutrons produced in a nuclear environment. The focus of this thesis is twofold. First, the study creates a human phantom capable of accurately representing the human body with organs of interest to include skin, bone, muscle, brain, and gastro-intestinal tract. Second, the study produces a user-ready version of the Oak Ridge National Laboratory (ORNL) MORSE computer code. MORSE is a Monte-Carlo method, neutron/gamma-ray transport code distributed by the Radiation Shielding Information Center (RSIC) at ORNL. The final modified version of MORSE should be capable of determining a normalized absorbed organ dose in a human phantom from any of five source configurations:

1. isotropic from a sphere surrounding the phantom,
2. mono-directional from the right side of the phantom (+x direction),
3. mono-directional from above the phantom (-z direction),
4. mono-directional from the front of the phantom (+y direction),
5. mono-directional from behind the phantom (-y direction).

Further, the final version should allow for a variety of source energy spectra to allow for various problems ranging from thermonuclear or fission weapons to simple laboratory neutron sources.

The code modification is limited to a standing phantom with five defined organs: skin, bone, brain, gastro-intestinal tract, and muscle (includes the remainder of the body tissue). Only these five organs are modeled since the

primary interest is an immediately incapacitating dose.

Approach

The ultimate goal of this project is reached by following six major steps:

1. development of a three-dimensional phantom detailed enough to provide accurate representation of the human body and the organs of interest,
2. preparation of the multi-group neutron cross sections required to model the phantom organs and surrounding air volume,
3. familiarization with the standard version of the MORSE computer code,
4. modification of MORSE subroutines to produce a phantom-problem specific code,
5. analysis and verification of the phantom problem for a fission weapon neutron spectrum,
6. final preparation of the user-ready version of MORSE.

The first and second steps are self explanatory. The third step is essentially debugging MORSE and running a sample problem included with the RSIC code package. The fourth step covers five stages in the evolution of the phantom version of MORSE using various geometric shapes to test the subroutine modifications. The modifications are detailed in the next chapter.

The fifth step is an exercise in producing results, testing the sensitivity of the results to variable parameters, and validating the results.

The final step is the preparation of the user-ready version of MORSE. The user-ready version is the goal of the thesis since it provides a means of easily determining normalized absorbed organ doses for a wide variety of problems.

Sequence of Presentation

The remaining pages of this thesis describe the stages of work in detail. Chapter II describes the phantom and cross section production followed by the evolution of the MORSE subroutines. Additionally, Chapter II includes some background about MORSE. Chapter III contains the results of this research along with a sample case example. The final chapter (IV) contains the conclusions of this thesis and recommendations for further research.

II. Preparation and Evolution of MORSE Files

Overview

The MORSE computer code package is a complex, and sometimes confusing, menagerie of FORTRAN subroutines and input data files. Analyzing a given problem may require modifications to many of the subroutines and data files -- a feat not easily performed without understanding MORSE and the Monte Carlo method.

This chapter details the steps performed to modify MORSE to meet the goals of this thesis. First, some background about MORSE is presented followed by sections on the preparation of the human phantom and neutron cross sections. The last portion of the chapter includes the 10 stages involved in the evolution of the final MORSE version.

MORSE Background

The MORSE (Multi-group Oak Ridge Stochastic Experiment) code is a multi-group neutron and gamma-ray transport code developed at ORNL (13:4.2-1). The code is based on Monte Carlo methods for determining probable particle interactions for a wide variety of problems in three-dimensional geometry. Solving problems in three-dimensional geometry is made possible by the combinatorial geometry (CG) module in MORSE. The CG module determines three-dimensional volumes and boundaries from combinations of various geometric shapes as discussed in the MORSE applications guide (5:21-28).

MORSE uses Monte Carlo probability distribution functions to determine fluence estimates from a user specified sample population. MORSE tracks the entire population from particle "birth" (source production) through particle "death" (escape from geometry, variance reduction kill, or time kill, all discussed

below) assigning a statistical weight at each stage of the particle's life.

A particle birth includes a starting location, energy group, direction, and statistical pathlength. Along the pathlength a particle collides with host nuclei or crosses boundaries between two geometry zones. Following each collision, MORSE multiplies the particle's weight by Σ_s/Σ_t (Σ_s and Σ_t are the scattering and total macroscopic cross section, respectively) to account for the probable fraction lost to absorption. Additionally, MORSE determines a new energy and direction following each collision.

Particles crossing a boundary into a defined region either undergo collisions or cross another boundary. Particles leaving (escaping from) the defined boundary are "killed" by reducing their weight to zero. Figure 1 is a simplified flowchart of a particle history (referred to as "random walk").

Unfortunately, if a particle does not leave the geometry it continues to "bounce" around to very low energies with little or no contribution to an overall fluence estimate. The many low energy collisions can add substantial computational time to the overall problem. On the other hand, if particles escape from the system at high energies, smaller geometry zones may not receive enough tracklengths to produce good statistics. Time and variance reduction techniques are available with MORSE to effectively solve both problems. Time kill, Russian roulette, and splitting are examples of such techniques and are described below.

The time kill technique is simply a particle age limit determined by the user. Throughout the random walk MORSE keeps track of each particle's age. When the age of a particle is longer than the user-specified time kill value, the particle's weight is set to zero.

Russian roulette is also useful for the first problem above. The technique determines whether particles of low weight are allowed to continue the random walk. Implementation of Russian roulette requires a minimum stipulated weight

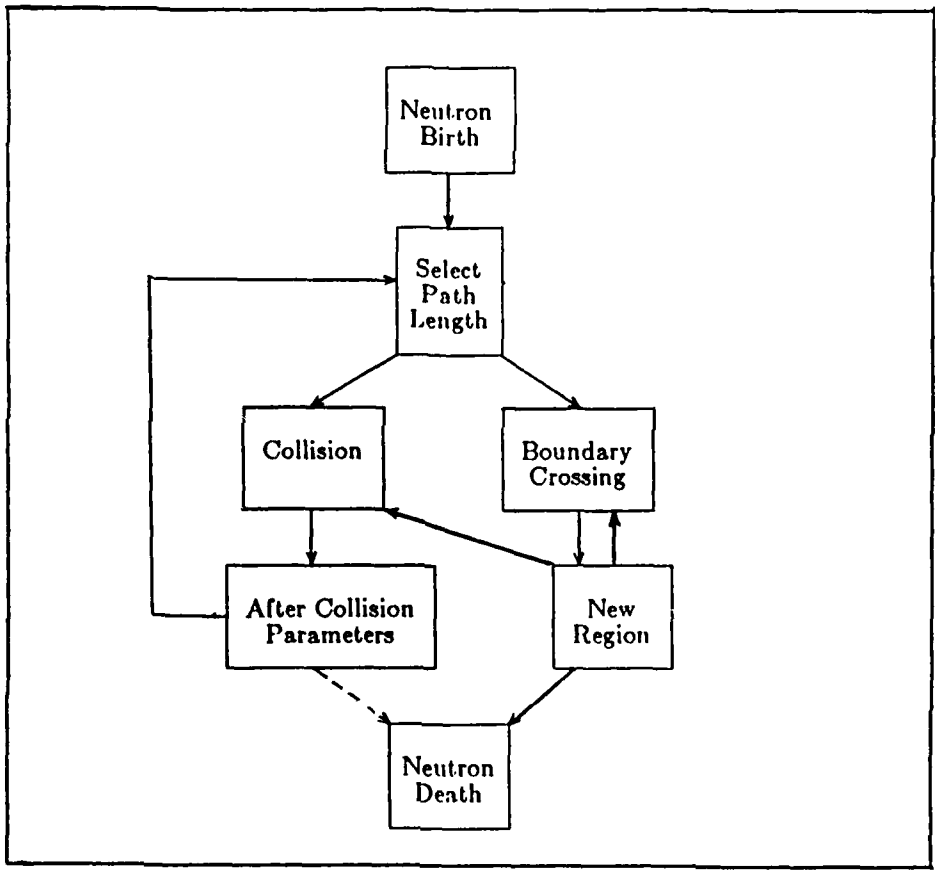

Figure 1. Random Walk Flow-Chart

value MORSE uses to determine which particles undergo the technique. Additionally, MORSE requires an average weight value to assign to particles surviving Russian roulette. The chance of survival is the ratio of the lower weight value to the average weight value (16:294).

Splitting is a useful technique to increase the number of particles in regions more than a few mean free paths from the surface of the geometry. Within a user-specified region, MORSE splits a particle if the weight is greater than a

specified minimum value. The first step is multiplication of the particle weight by Σ_s/Σ_t to account for the fraction lost to absorption. Next, MORSE divides the weight by two to account for the two particles produced. The splitting continues until the particle weights fall below the minimum value mentioned above. This method produces many more statistical particles possibly resulting in a reduction of sample variance (4:16).

In order to use MORSE for a specific problem, problem-specific input data and problem dependent user-modified subroutines must be produced. The input data includes parameters controlling Monte-Carlo random walk, combinatorial geometry description, cross section information, and user-requested calculations. The user-modified subroutines (specific to this thesis) include routines to generate source neutrons, assign cross section media regions, determine dose contributions from single neutron events, and read in additional input data.

Human Phantom

The human phantom created in this project is 5 feet, 9.3 inches (176 cm) tall with a mass of 151 $\pm$ 2 pounds (69 $\pm$ 1 kg). The phantom is modeled from the ICRP reference man (10), Gray's anatomy(9), the Snyder phantom model (8:1473-1478), and discussions with a medical doctor (14). The reference man is 5 feet, 9 inches (175 cm) tall and weighs 154 pounds (70 kg). The mass of the phantom model differs from that of the reference man primarily because the phantom is somewhat more simplistic than the reference man. Figure 2 shows a three-dimensional representation of the model used in this study. The figure on the left represents the skin of the phantom. The figure on the right shows the phantom bones.

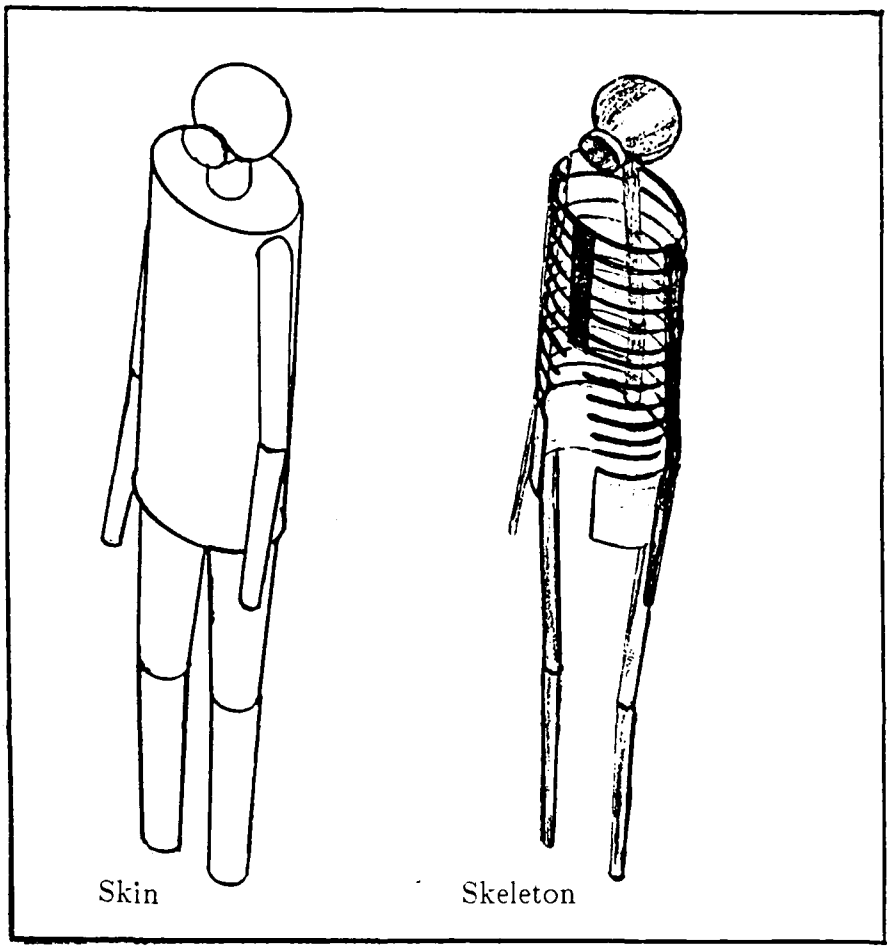

Figure 2. Three-Dimensional Representation of Human Phantom

The coordinate axis of the phantom originates at the center of the base of the right elliptical cylinder defining the torso. The positive x-direction is toward the left-hand side of the body, the positive y-direction is toward the back of the body, and the positive z-direction is toward the head. Sixty-one geometric shape entries are combined to produce the phantom with its five organs: skin, bone, brain, gastro-intestinal tract, and muscle (includes the remaining tissue of the body). Additional organs are not modeled since this thesis is (primarily interested in)

determining incapacitating doses to the body. The actual data lines that produce the phantom are included in the input data file listed in Appendix A. Figures 3 and 4 are computer generated phantom representations to verify correctness of the input. The first representation (Figure 3) is a cross sectional front view of the phantom cut down the center plane (y=0). The darker points represent bone, the smaller points represent muscle, the dashes are the gastro-intestinal tract, and the dark boxes represent the brain. The relatively small thickness of the skin does not allow for a computer representation on this scale. The second (Figure 4) is a cross sectional side view cut down the center plane (x=0). The representations of Figure 4 are the same as for Figure 3. Additionally, the skin in representable on this size view (shown as "+" signs). The representation in Figure 4 is somewhat "squashed" down but accurately depicts the relative position of each organ. Both representations are produced using an RSIC code written specifically to create such images (13:Sect 3).

The input data file in Appendix A contains the exact descriptions of the 62 geometric shapes required to produce the phantom. The individual depictions of each phantom component are included in the thesis notebook (1). The phantom model consists of four main regions: head, torso, legs, and arms.

Head. The head is a sphere of outer radius 9.0 cm, with a right circular cylinder of radius 7.5 cm attached to form the jaw. Another right circular cylinder of radius 5.2 cm attaches from below the sphere to form the neck. The skin covers the head with a thickness of 0.2 cm (same thickness over the entire phantom). The skull is 0.5 cm in thickness adjacent to the skin of the head. The spinal cord begins at the base of the skull and extends to the torso as described in the next paragraph. The brain is approximately two-thirds of a sphere of radius 8.3 cm within the spherical portion of the skull. The remaining space within the head is muscle.

Torso. The torso is a right elliptical cylinder 70 cm tall with a minor radius of 10.5 cm and a major radius of 15.5 cm. Beneath the skin lies the ribcage, spinal cord, pelvis, and gastro-intestinal tract. The ribcage is a right elliptical cylindrical shell cut by cross sectional planes to produce 11 ribs 0.5 cm deep and 1.7 cm high (see Figure 2). The lower five ribs are cut by wedges to produce an abdominal opening. Additionally, a 0.5 cm thick sternum is over the top six ribs.

The spinal cord is a 2.0 cm thick right circular cylinder originating beneath the skull and angling toward the middle portion of the ribcage. From there, the spinal cord follows the ribcage straight down to the pelvis. The pelvis is a right elliptical cylindrical shell with a portion cut off in front (see Figure 2). The pelvis originates at the base of the torso and is 19.6 cm high. The gastro-intestinal tract is a right elliptical cylinder originating inside and approximately half way up the pelvis. The tract extends an additional 5.2 cm above the pelvis. All additional space within the torso is muscle.

Legs. The legs are divided into upper and lower legs, both truncated right cones. The upper and lower radii of the entire leg are 8.5 cm and 3.5 cm, respectively. Within each leg are the upper and lower leg bones. The upper leg bones are right circular cylinders of radius 2.5 cm. The bones originate at the outer base of the pelvis and extend to the center location between the upper and lower legs. The lower leg bones (tibia and fibula) are modeled as a single right circular cylinder within each leg.

Arms. The arms are also divided into upper and lower sections. The radii of the upper and lower portions of the arms are 4.0 and 2.0 cm, respectively. The arm bones are right circular cylinders of 1.5 cm (upper arm) and 1.25 cm (lower arm - ulna and radius combined) through the center of each arm.

All of the organs, with exception of the bone, are approximated as homogeneous tissue with a density of 0.987 g/cm^3 (10:290-324). Although this may seem to

be a crude approximation, actual organ densities are very difficult, if not impossible, to determine. Throughout the day organs vary in composition because of typical body functions. However, the average daily mass to volume ratio of most organs is almost exactly that of tissue (14). The bone is modeled with a density of 1.49 g/cm^3. The constituents of the phantom tissue and bone are listed in Table I. Table II presents the phantom organ volumes and their uncertainty in measurement. The uncertainties result since the more complex geometric combinations produce non-trivial volume equations.

Cross Section Mixing

The MORSE computer package includes a choice for mixing cross sections: mix during each problem analysis or pre-mix prior to problem analysis. The phantom problem requires the latter because of buffer storage problems resulting from the large storage requirements of the geometry and particle histories.

Three media are required for this problem: soft tissue, bone, and air. Table I contains the elemental constituents of soft tissue and bone. The air is a mixture of nitrogen (80.0 percent by weight) and oxygen (20.0 percent by weight). The individual elemental cross sections are from the RSIC DLC-37 data library containing 37 neutron and 21 gamma-ray energy group cross sections (2). The library is an ANISN format library with the following cross sections: absorption, fission, total, within group, and group-to-group. The scattering coefficients are represented by a P3 Legendre polynomial expansion of the differential scattering cross sections.

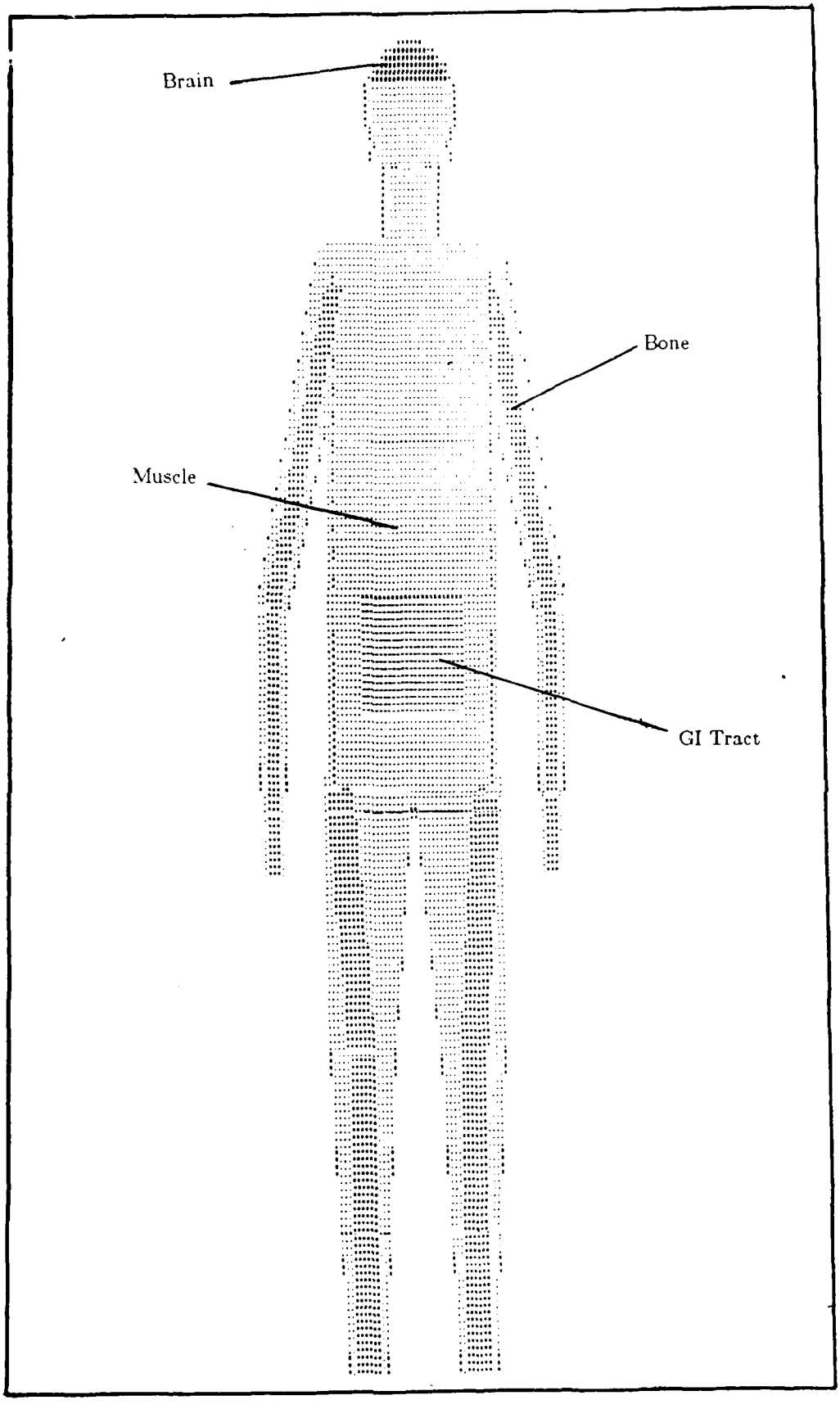

Figure 3. MORSE cross sectional Representation of Phantom from Front

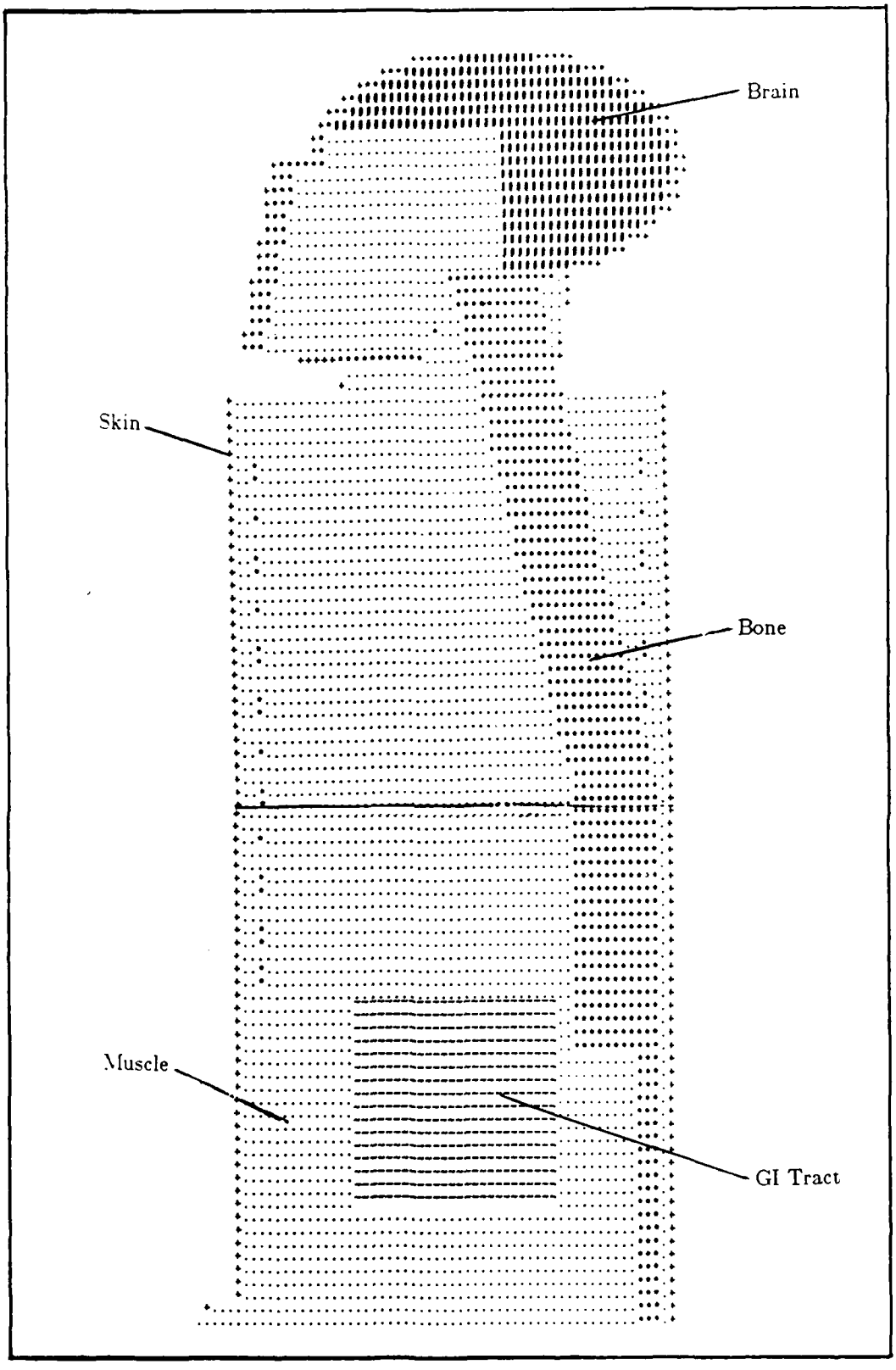

Figure 4. MORSE cross sectional Representation of Phantom from Side

Table I

Elemental Composition and Atom Densities
of Phantom Organs

Element	Weight Percent		Atom Dens. (atoms/b-cm)	
	Tissue	Bone	Tissue	Bone
H	10.47	7.04	.06174	.06267
C	23.02	22.79	.01139	.01703
N	2.34	3.87	.000993	.002479
O	63.21	48.56	.02348	.02723
Na	0.13	0.32	.000080	.000125
Mg	0.02	0.11	.000004	.000041
P	0.24	6.94	.000046	.002010
S	0.22	0.17	.000041	.000048
Cl	0.14	0.14	.000024	.000035
K	0.21	0.15	.000032	.000034
Ca	0.00	9.91	0.0	.002219

Table II

Phantom Organ Volumes

Organ	Volume(cm)	Uncertainty (%)
Skin	3340	1.9
Bone	6700	2.7
Muscle	50000	<0.01
Brain	1555	0.0
GI Tract	2120	0.0

Evolution of Final MORSE Version

The final version of MORSE determines normalized absorbed radiation dose to a human phantom for any of the five source descriptions: isotropic and mono-directional from right side, above, front, and back. By simply modifying the input data file, a wide range of problems can be solved without any modifications to MORSE. Appendix A contains an example input data file. The remainder of this section covers the 10 stages of MORSE evolution resulting in the final MORSE version of this thesis.

Familiarization. The MORSE code package at the Air Force Institute of Technology (AFIT) is a version modified to run on a Digital Electronics Corporation VAX/VMS system. Subsequently, none of the many FORTRAN files require modification if the code is run on the AFIT CSC computer system. However, as a check to ensure program integrity, the code includes sample problems complete with results. The results of the first sample problem (13: Sect II,2-1) run at AFIT did indeed match those forwarded by RSIC.

Initial Test. This stage covers the initial modification of MORSE subroutines and a comparison with diffusion equation results. The problem is a sphere of water with a 15 cm radius. The sphere receives an isotropic neutron fluence of $1\ n/cm^2$ at the sphere surface (directed over 2π steradians inward). The cross sections are for thermal energy neutrons in water but are used over a range of 1 Mev to 0.025 ev in this 1-group problem. The initial test includes rewriting the fluence estimator subroutines to include a track length estimator for spherical geometry. The track length estimator equation is:

$$\phi = \frac{\sum_{i=1}^{N} l_i}{N\ V} \qquad (1)$$

where ϕ is the normalized fluence in n/cm^2/source n, l_i is the distance traveled (tracklength) by particle i in the sphere (cm), N is the number of particles tracked by MORSE, and V is the volume of the sphere (cm^3). MORSE models the incident neutron fluence on the sphere surface using random number generators to direct the neutrons isotropically inward. MORSE also determines whether or a not a neutron is within the sphere volume using a simple test: is the distance between the neutron and the center of the sphere greater than the radius of the sphere? If the distance is larger, the neutron's tracklength is not included in the fluence estimate. A sample population of 5000 source neutrons produces a fluence estimate in the sphere volume of 4.0 X $10^{-4} n/cm^2/source\ n$ with a standard deviation of 5.0%.

The solution to the diffusion equation (1:29) for the water sphere problem is:

$$\phi(r) = \frac{0.71 sinh(Br)}{r} \tag{2}$$

where $\phi(r)$ is the fluence at radius r (n/cm^2), and B is the material buckling (cm^{-1}). The diffusion equation solution is for a non-multiplying medium using one-group thermal neutron cross sections. Integrating eq. (2) over the sphere's radius and dividing by the volume produces a volume average fluence over the 15 cm radius sphere of 0.37 n/cm^2 (1:32). In order to compare solutions between MORSE and the diffusion equation, the MORSE results are multiplied by the total number of source neutrons from a normalized fluence of 1 n/cm^2 on the sphere surface ($4\pi(15cm)^2 * 1n/cm^2 = 2830$ source neutrons) to produce 1.1 n/cm^2. However, the MORSE results are still a factor of two larger since the source model was only over 2π and not 4π steradians. The final MORSE result is then 0.55 $n/cm^2 \pm 0.03$. The MORSE and diffusion equation results are within an order of magnitude of each other.

Multiple Volume Test. The next MORSE modification solves the previous problem with a slight variation: the source is on a spherical surface with a radius of 35 cm, concentric to the original sphere. This modification is a benchmark for future geometry modifications. The MORSE analysis of the 2000 neutrons directed isotropically inward results in a normalized fluence in the inner sphere of $4.4\ X\ 10^{-5} n/cm^2/source\ n$ with a standard deviation of 7.0%.

Non-Spherical Geometry. The modifications prior to this stage are strictly for concentric sphere problems. The modification of this stage is to determine fluence in any three-dimensional shape using geometry independent tests. The new tests determine the current region of the neutron location. A region is a user-defined volume described in the combinatorial geometry section of the input data file. Using the same problem as in the multiple volume test but with the new subroutine modification, MORSE again produces a normalized fluence of $4.4\ X\ 10^{-5} n/cm^2/source\ n$. The result validates the geometry independent modification in the spherical problem.

Non-Spherical Test. This stage determines the validity of the previous modification on non-spherical geometry. The problem consists of two cubes, full of vacuum, as shown in Figure 5. Neutrons are incident on the left-hand surface of cube 1 in the $+x$ direction. The only modification to MORSE is a statement after each source neutron is created to print the starting location. Knowing the starting location determines whether a neutron travels only through cube 1 or both cube 1 and cube 2. Since the cubes are full of vacuum, the neutrons do not collide and change direction. The code produces normalized fluences of $0.250 n/cm^2/source\ n$ and $0.0875 n/cm^2/source\ n$ for cube 1 and cube 2, respectively, from 20 starting source neutrons. Hand calculation (1:39) verifies these results since all 20 must pass through cube 1 while only 7 pass through cube 2.

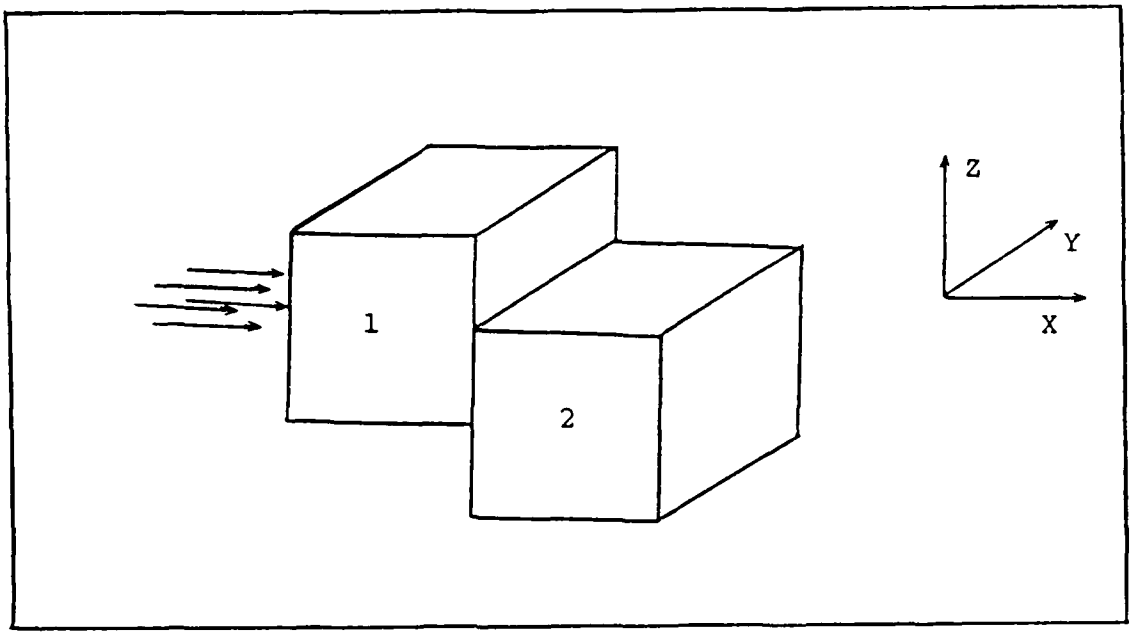

Figure 5. Geometry for Non-Spherical Test.

Multiple-Group Upgrade. The next stage is the conversion from a single group water problem to a 37 group tissue problem using the geometry defined in Stage 2 (source on surface of 15 cm radius sphere). An analysis of 2000 neutrons produces a normalized fluence in the sphere of 1.52 X $10^{-3} n/cm^2$/source n with a standard deviation of 5.3%. This fluence is approximately three times larger than the fluence produced using one-group cross sections; the difference is attributable to the ability of each neutron to scatter from group to group, with varying cross sections. between 1 Mev and 0.025 ev (group 21 to group 37).

Phantom Geometry Addition. The seventh stage is the first to use the phantom geometry. The main purpose of this stage is to determine fluence sensitivity to change in organ volume. Table II shows three of the five organs (skin. bone. and muscle) have an uncertainty in volume measurement. To determine the

sensitivity of the estimated fluence to the volume uncertainty, five analyses are run through MORSE. In each case, only the organ volumes are changed in the input data file since the dose effect from individual tracklengths is divided by the volume (eq. 1). Table III shows the results of the five 1000-neutron analyses.

Table III

Fluence Sensitivity to Organ Volume Uncertainty

Analysis #	Normalized Fluence [n/cm^2/source n]		
	Skin	Bone	Muscle
1	9.46E-6	1.16E-5	9.94E-6
2	9.30E-6	1.16E-5	9.94E-6
3	9.46E-6	1.12E-5	9.94E-6
4	9.46E-6	1.16E-5	9.85E-6
5	9.30E-6	1.12E-5	9.85E-6

The first analysis uses the mean phantom organ volumes shown in Table II. The second analysis uses the upper limit of the skin volume and leaves the bone and muscle unchanged. The third analysis uses the upper limit of the bone volume and mean volumes of skin and muscle. The fourth analysis uses the upper limit of the muscle and mean volumes of the skin and bone. The fifth analysis uses the upper limit of each organ volume. The results show that the skin's volume uncertainty varies the fluence by no more than 1.7%; the bone volume uncertainty varies the fluence by no more than 3.4%; and the muscle volume uncertainty varies the fluence by no more than 0.9%. The fifth analysis shows that whether the organ volumes are varied singly or all at once, their net effect is the same: a variation of a specific organ volume only affects that organ's fluence. The remaining two organs have no volume uncertainties and do not show a variation in fluence results from the above analyses.

Absorbed Dose Calculation. The next stage is the conversion of MORSE output from normalized fluence to normalized absorbed dose. The conversion method recommended by the International Commission on Radiation Units and Measurements (15:95) is use of KERMA factors. KERMA stands for Kinetic Energy Released in MAterial and is the sum of initial kinetic energies, of charged particles resulting from neutron interaction in a volume, per mass of material in the volume. KERMA factors are actually a KERMA to unit fluence ratio. The KERMA factors used in this project are from the RSIC cross section library mentioned above in the cross section mixing section. Individual elemental KERMA for the 11 constituent elements of tissue and bone are mixed to produce tissue and bone KERMA factors for all 37 energy groups. The tissue and bone KERMA factors are listed in Table IV (energy ranges of the groups are listed in Table V).

The normalized dose for organ m is then determined as:

$$\hat{D}_i^m = \hat{\phi}_i^m * KF_i^m \quad [RAD-(tissue\ or\ bone)/(source\ n)]. \tag{3}$$

where $\hat{\phi}_i^m$ is the MORSE estimate of normalized fluence (Eq. 1) of energy group i and KF_i^m is the KERMA factor of the i^{th} energy group.

A total absorbed dose is calculated by multiplying the MORSE output of normalized dose with the total incident fluence:

$$D^m = \hat{D}^m * \phi_{inc} \tag{4}$$

where D^m is the total absorbed dose of organ m and ϕ_{inc} is the incident neutron fluence.

Sample Analysis. The ninth stage of the MORSE evolution is analysis of the phantom using a fission weapon energy spectrum (2:11) with neutrons incident five different ways:

1) isotropic from a sphere surrounding the phantom (radius = 100 cm, center at 0,0,10),

2) mono-directional in the +x direction (incident on the right side of the phantom from a source plane defined by $x = -30$, $-13 \leq y \leq 13$, $-80 \leq z \leq 96$),

3) mono-directional in the -z direction (incident from above the phantom from a source plane defined by $-30 \leq x \leq 30$, $-13 \leq y \leq 13$, $z = 96$),

4) mono-directional in the +y direction (incident on the front of the phantom from a source plane defined by $-30 \leq x \leq 30$, $y = -13$, $-80 \leq z \leq 96$),

5) mono-directional in the -y direction (incident on the back side of the phantom from a source plane defined by $-30 \leq x \leq 30$, $y = 13$, $-80 \leq z \leq 96$.

The results are presented in Chapter III, Results. Table V shows the energy distribution of the weapon spectrum.

Final MORSE Version. The final stage is the creation of the user-ready version of MORSE. Prior to this stage, using different source directions required modification to the MORSE FORTRAN subroutines. Instead, MORSE is modified to read special input data describing the source direction required. Additionally, the surface area of the respective source type is multiplied with the dose per source neutron to produce dose per unit fluence (normalized dose) in units of GRAY-(tissue or bone)/(n/cm^2). Appendix B contains the final modified MORSE subroutines. The results of this stage are presented in Chapter III of this thesis.

Table IV.

Tissue and Bone KERMA Factors

Energy Group	KERMA Factors	
	Tissue [RAD-Tiss/(n/cm²)]	Bone [RAD-Bone/(n/cm²)]
1	0.723×10^{-08}	0.551×10^{-08}
2	0.686×10^{-08}	0.520×10^{-08}
3	0.664×10^{-08}	0.502×10^{-08}
4	0.653×10^{-08}	0.493×10^{-08}
5	0.638×10^{-08}	0.482×10^{-08}
6	0.619×10^{-08}	0.468×10^{-08}
7	0.612×10^{-08}	0.461×10^{-08}
8	0.586×10^{-08}	0.440×10^{-08}
9	0.569×10^{-08}	0.427×10^{-08}
10	0.539×10^{-08}	0.399×10^{-08}
11	0.528×10^{-08}	0.391×10^{-08}
12	0.504×10^{-08}	0.366×10^{-08}
13	0.472×10^{-08}	0.341×10^{-08}
14	0.448×10^{-08}	0.320×10^{-08}
15	0.441×10^{-08}	0.316×10^{-08}
16	0.419×10^{-08}	0.300×10^{-08}
17	0.356×10^{-08}	0.249×10^{-08}
18	0.326×10^{-08}	0.226×10^{-08}
19	0.320×10^{-08}	0.221×10^{-08}
20	0.276×10^{-08}	0.191×10^{-08}
21	0.210×10^{-08}	0.145×10^{-08}
22	0.130×10^{-08}	0.894×10^{-09}
23	0.801×10^{-09}	0.547×10^{-09}
24	0.559×10^{-09}	0.381×10^{-09}
25	0.315×10^{-09}	0.214×10^{-09}
26	0.215×10^{-09}	0.146×10^{-09}
27	0.147×10^{-09}	0.100×10^{-09}
28	0.627×10^{-10}	0.428×10^{-10}
29	0.221×10^{-10}	0.152×10^{-10}
30	0.919×10^{-11}	0.645×10^{-11}
31	0.310×10^{-11}	0.241×10^{-11}
32	0.103×10^{-11}	0.120×10^{-11}
33	0.919×10^{-12}	0.141×10^{-11}
34	0.134×10^{-11}	0.222×10^{-11}
35	0.225×10^{-11}	0.375×10^{-11}
36	0.369×10^{-11}	0.610×10^{-11}
37	0.151×10^{-10}	0.247×10^{-10}

Table V.

Fission Weapon Neutron Energy Spectrum

Energy Group	Upper Energy Limit (Mev)	Fraction
1	$1.96x10^{+07}$	0.000
2	$1.69x10^{+07}$	0.000
3	$1.49x10^{+07}$	0.000
4	$1.42x10^{+07}$	0.000
5	$1.38x10^{+07}$	0.000
6	$1.28x10^{+07}$	0.000
7	$1.22x10^{+07}$	0.000
8	$1.11x10^{+07}$	0.000
9	$1.00x10^{+07}$	0.004
10	$9.05x10^{+06}$	0.004
11	$8.19x10^{+06}$	0.005
12	$7.41x10^{+06}$	0.007
13	$6.38x10^{+06}$	0.018
14	$4.97x10^{+06}$	0.003
15	$4.72x10^{+06}$	0.009
16	$4.07x10^{+06}$	0.055
17	$3.01x10^{+06}$	0.032
18	$2.39x10^{+06}$	0.011
19	$2.31x10^{+06}$	0.097
20	$1.83x10^{+06}$	0.147
21	$1.11x10^{+06}$	0.216
22	$5.55x10^{+05}$	0.150
23	$1.58x10^{+05}$	0.019
24	$1.11x10^{+05}$	0.121
25	$5.25x10^{+04}$	0.057
26	$2.48x10^{+04}$	0.006
27	$2.19x10^{+04}$	0.024
28	$1.03x10^{+04}$	0.014
29	$3.35x10^{+03}$	0.000
30	$1.23x10^{+03}$	0.000
31	$5.83x10^{+02}$	0.000
32	$1.01x10^{+02}$	0.000
33	$2.90x10^{+01}$	0.000
34	$1.07x10^{+01}$	0.000
35	$3.06x10^{+00}$	0.000
36	$1.13x10^{+00}$	0.000
37	$4.14x10^{-01}$	0.000

III. Results

Overview

The user-ready version of MORSE produced from the last stage of the code evolution is the version used for analysis in this chapter. The results are by no means for an actual problem but instead show how MORSE is used, in its modified form, to determine absorbed neutron dose to organs of a phantom.

This chapter contains four results sections; analyses using the user-ready version of MORSE, a comparison of MORSE results to results from another study, variance and processing time reduction analyses, and a sample calculation using MORSE output to determine estimated dose to a soldier. In all cases, the analyses consist of neutrons incident on the phantom in each of the five source configurations described in the Sample Analysis Stage of Chapter II. The energy spectrum is the fission weapon spectrum shown in Table V. Because of the buffer storage problem described in Chapter II, MORSE is capable of analyzing only 1200 particles per batch in problems with no splitting and 1000 particles per batch in problems with splitting.

Test Case Results.

The first analysis (referred to as "test case") uses a sample population of 24,000 neutrons (20 batches of 1200 neutrons per batch). The results are presented in Table VI. In each box of Table VI, the top number is the normalized (unit) dose and the bottom number is the fractional standard deviation. The table shows the form of results attainable for a given neutron energy spectrum. Multiplication of any of the organ-source direction results with an incident fluence provides a total absorbed dose to the organ (eq. 4). A solution set such as Table VI eliminates the need for any future computer analysis for a given energy spectrum. Note the large central processing unit (CPU) times required for the mono-

directional sources compared with the isotropic source. These larger CPU times occur since many more source neutrons interact with the phantom than in the isotropic case (see source plane descriptions in the Sample Analysis Stage of Chapter II).

Table VI.

Normalized Dose Results - Test Case

Organ #	Norm. Dose [Gray/(n/cm^2)] from given source				
	ISOTROPIC	Rt. Side	Above	Front	Back
Skin	$6.99x10^{-12}$ 0.05	$1.09x10^{-11}$ 0.01	$3.53x10^{-12}$ 0.02	$1.47x10^{-11}$ 0.02	$1.49x10^{-11}$ 0.02
Bone	$3.66x10^{-12}$ 0.07	$5.40x10^{-12}$ 0.02	$1.19x10^{-12}$ 0.02	$7.36x10^{-12}$ 0.03	$9.41x10^{-12}$ 0.02
Muscle	$4.27x10^{-12}$ 0.06	$6.43x10^{-12}$ 0.01	$1.22x10^{-12}$ 0.03	$1.17x10^{-11}$ 0.02	$1.11x10^{-11}$ 0.01
Brain	$4.91x10^{-12}$ 0.15	$8.48x10^{-12}$ 0.05	$1.12x10^{-11}$ 0.02	$5.42x10^{-12}$ 0.08	$1.17x10^{-11}$ 0.08
GI Tract	$1.22x10^{-12}$ 0.39	$2.66x10^{-12}$ 0.15	$4.16x10^{-15}$ 0.87	$6.66x10^{-12}$ 0.08	$6.76x10^{-12}$ 0.10
CPU (min)	12.70	84.76	46.43	80.90	79.39

Note: the numbers below normalized dose are fractional standard deviation.

Comparison of Results.

The vast majority of open literature on neutron doses addresses results in terms of transmission factors. A transmission factor is the ratio of KERMA in an organ to soft-tissue KERMA in free air (no organ present). The organ KERMA for the phantom problem is simply the MORSE output, since the output is a normalized fluence: $KERMA = GRAY/(n/cm^2) * 1\ n/cm^2 = GRAY$. The KERMA in free air (more commonly referred to as "free-in-air KERMA") for the phantom problem is determined (1:58) as:

$$K^i_{fia} = KF^i_{tissue} * f^i \tag{5}$$

which leads to

$$K_{fia} = \sum_{i=1}^{37} K^i_{fia} \tag{6}$$

where KF^i_{tissue} is the tissue KERMA factor of the i^{th} energy group and f^i is the source fission energy fraction of the i^{th} group. Table VII lists the free-in-air KERMA values for all 37 neutron energy groups. The total free-in-air tissue KERMA is 2.1 X 10^{-9} RAD-tissue.

The only two organs applicable for comparison with results from other studies are the brain and skeleton. This is primarily because the remaining organs of the human phantom studied here are not commonly analyzed by others (this study is primarily interested in immediate incapacitation to organs). Table VIII shows a comparison of the MORSE bone and brain phantom transmission factors to transmission factors of Japanese atomic bomb survivors (6:617). The Japanese survivor data is from a joint Science Applications International Corporation (SAIC) and RSIC data base.

Table VII

Free-in-Air KERMA Values

Energy Group	Fraction	KERMA Factor [RAD-Tis./(n/cm^2)]	KERMA [RAD-Tis./(n/cm^2)]
1	0.0000	$0.723x10^{-8}$	0.0
2	0.0000	$0.686x10^{-8}$	0.0
3	0.0000	$0.664x10^{-8}$	0.0
4	0.0000	$0.653x10^{-8}$	0.0
5	0.0000	$0.638x10^{-8}$	0.0
6	0.0000	$0.619x10^{-8}$	0.0
7	0.0000	$0.612x10^{-8}$	0.0
8	0.0000	$0.586x10^{-8}$	0.0
9	0.0038	$0.569x10^{-8}$	$0.219x10^{-10}$
10	0.0035	$0.539x10^{-8}$	$0.189x10^{-10}$
11	0.0054	$0.528x10^{-8}$	$0.285x10^{-10}$
12	0.0074	$0.504x10^{-8}$	$0.370x10^{-10}$
13	0.0184	$0.472x10^{-8}$	$0.869x10^{-10}$
14	0.0033	$0.448x10^{-8}$	$0.146x10^{-10}$
15	0.0085	$0.441x10^{-8}$	$0.374x10^{-10}$
16	0.0750	$0.419x10^{-8}$	$0.314x10^{-9}$
17	0.0324	$0.356x10^{-8}$	$0.115x10^{-9}$
18	0.0106	$0.326x10^{-8}$	$0.346x10^{-10}$
19	0.0972	$0.320x10^{-8}$	$0.311x10^{-9}$
20	0.1470	$0.276x10^{-8}$	$0.406x10^{-9}$
21	0.2160	$0.210x10^{-8}$	$0.453x10^{-9}$
22	0.1500	$0.130x10^{-8}$	$0.195x10^{-9}$
23	0.0193	$0.801x10^{-9}$	$0.155x10^{-10}$
24	0.1210	$0.559x10^{-9}$	$0.676x10^{-10}$
25	0.0573	$0.315x10^{-9}$	$0.180x10^{-10}$
26	0.0060	$0.215x10^{-9}$	$0.129x10^{-11}$
27	0.0240	$0.147x10^{-9}$	$0.353x10^{-11}$
28	0.0144	$0.627x10^{-10}$	$0.903x10^{-12}$
29	0.0000	$0.221x10^{-10}$	0.0
30	0.0000	$0.919x10^{-11}$	0.0
31	0.0000	$0.310x10^{-11}$	0.0
32	0.0000	$0.103x10^{-11}$	0.0
33	0.0000	$0.919x10^{-12}$	0.0
34	0.0000	$0.134x10^{-11}$	0.0
35	0.0000	$0.225x10^{-11}$	0.0
36	0.0000	$0.369x10^{-11}$	0.0
37	0.0000	$0.151x10^{-10}$	0.0
Total			$2.100x10^{-9}$

Table VIII

KERMA Transmission Factors: Phantom versus Japanese Survivors

	MORSE	SAIC/ORNL
Bone	0.37 (0.03)	0.36 (0.05)
Brain	0.31 (0.12)	0.37 (0.05)

The numbers in parentheses are fractional standard deviations.

The transmission factor comparison is very close for the bone results. The brain comparison is not as good and therefore prompted another look at the modeling of the head (from this study). The size of the brain from the SAIC/ORNL model is not known for comparison, but the phantom brain from this study may be approximately 100 cm^3 too large if the Snyder model is accurate (8:1476). As a check, a reduction in the size of the phantom brain by $100cm^3$ produces a transmission factor of 0.32. Any further discrepancies in the transmission factors cannot be fully realized without knowing the details of the SAIC/ORNL model. One can assume the skull thickness (uniform in this study) attenuates too many neutrons resulting in a lower transmission factor.

Time/Variance Reduction Results.

This third section of results contains four attempts to reduce sample variance and/or CPU time. The first series uses splitting, Russian roulette, and time kill for 24,000 source neutrons (24 batches of 1000). Time kill is a function that creates a neutron "death" for neutrons older than 10 microseconds. Ten microseconds is chosen to ensure the neutrons reach the lowest energy group before being eliminated. The results are presented in Table IX.

The second series uses splitting and time kill, again for 24 batches of 1000 neutrons. The results are presented in Table X.

The results of the third series are presented in Table XI. The series uses only time kill for 20 batches of 1200 particles.

The fourth and final series uses only time kill but runs 40 batches of 1200 neutrons. The results are presented in Table XII.

Table IX

Normalized Dose Results - Series #1

Organ #	Norm. Dose [Gray/(n/cm^2)] from given source				
	ISOTROPIC	Rt. Side	Above	Front	Back
Skin	$6.93x10^{-12}$ 0.05	$1.06x10^{-11}$ 0.02	$3.48x10^{-12}$ 0.02	$1.46x10^{-11}$ 0.01	$1.54x10^{-11}$ 0.02
Bone	$3.39x10^{-12}$ 0.06	$5.13x10^{-12}$ 0.02	$1.19x10^{-12}$ 0.03	$7.34x10^{-12}$ 0.02	$9.92x10^{-12}$ 0.02
Muscle	$4.26x10^{-12}$ 0.05	$5.13x10^{-12}$ 0.02	$1.25x10^{-12}$ 0.02	$1.12x10^{-11}$ 0.02	$1.10x10^{-11}$ 0.02
Brain	$5.45x10^{-12}$ 0.17	$9.58x10^{-12}$ 0.05	$1.07x10^{-11}$ 0.03	$5.64x10^{-12}$ 0.13	$9.95x10^{-12}$ 0.10
GI Tract	$1.42x10^{-12}$ 0.31	$2.35x10^{-12}$ 0.11	$2.73x10^{-14}$ 0.49	$7.59x10^{-12}$ 0.08	$6.03x10^{-12}$ 0.07
CPU (min)	11.55	63.69	12.57	71.96	69.07

Table X

Normalized Dose Results - Series #2

Organ #	Norm. Dose [Gray $/(n/cm^2)$] from given source				
	ISOTROPIC	Rt. Side	Above	Front	Back
Skin	$6.60x10^{-12}$ 0.045	$1.04x10^{-11}$ 0.013	$3.52x10^{-12}$ 0.024	$1.47x10^{-11}$ 0.012	$1.47x10^{-11}$ 0.013
Bone	$3.60x10^{-12}$ 0.081	$5.36x10^{-12}$ 0.022	$1.20x10^{-12}$ 0.023	$7.58x10^{-12}$ 0.023	$9.46x10^{-12}$ 0.016
Muscle	$3.96x10^{-12}$ 0.044	$6.22x10^{-12}$ 0.011	$1.24x10^{-12}$ 0.022	$1.12x10^{-11}$ 0.013	$1.08x10^{-11}$ 0.015
Brain	$5.07x10^{-12}$ 0.234	$8.68x10^{-12}$ 0.061	$1.03x10^{-11}$ 0.039	$6.33x10^{-12}$ 0.121	$1.255x10^{-11}$ 0.082
GI Tract	$1.28x10^{-12}$ 0.310	$2.36x10^{-12}$ 0.118	$1.41x10^{-14}$ 0.668	$5.91x10^{-12}$ 0.109	$7.34x10^{-12}$ 0.082
CPU (min)	5.73	14.96	11.73	12.79	12.43

Note: the numbers below normalized dose are fractional standard deviation.

Table XI

Normalized Dose Results - Series #3

Organ #	Norm. Dose [Gray $/(n/cm^2)$] from given source				
	ISOTROPIC	Rt. Side	Above	Front	Back
Skin	$6.94x10^{-12}$ 0.046	$1.07x10^{-11}$ 0.016	$3.46x10^{-12}$ 0.017	$1.45x10^{-11}$ 0.014	$1.50x10^{-11}$ 0.018
Bone	$3.54x10^{-12}$ 0.078	$5.51x10^{-12}$ 0.020	$1.22x10^{-12}$ 0.028	$7.47x10^{-12}$ 0.023	$9.54x10^{-12}$ 0.018
Muscle	$4.02x10^{-12}$ 0.050	$6.43x10^{-12}$ 0.016	$1.24x10^{-12}$ 0.020	$1.13x10^{-11}$ 0.015	$1.09x10^{-11}$ 0.013
Brain	$6.90x10^{-12}$ 0.172	$9.84x10^{-12}$ 0.069	$1.08x10^{-11}$ 0.035	$6.55x10^{-12}$ 0.148	$1.24x10^{-11}$ 0.054
GI Tract	$1.55x10^{-12}$ 0.401	$2.89x10^{-12}$ 0.096	$2.90x10^{-13}$ 0.747	$6.56x10^{-12}$ 0.096	$7.61x10^{-12}$ 0.083
CPU (min)	5.53	13.96	11.05	11.92	12.00

Note: the numbers below normalized dose are fractional standard deviation.

Table XII

Normalized Dose Results - Series #4

Organ #	Norm. Dose [Gray/(n/cm^2)] from given source				
	ISOTROPIC	Rt. Side	Above	Front	Back
Skin	$6.80x10^{-12}$ 0.035	$1.08x10^{-11}$ 0.011	$3.50x10^{-12}$ 0.013	$1.47x10^{-11}$ 0.011	$1.50x10^{-11}$ 0.011
Bone	$3.35x10^{-12}$ 0.053	$5.47x10^{-12}$ 0.013	$1.24x10^{-12}$ 0.018	$7.60x10^{-12}$ 0.016	$9.48x10^{-12}$ 0.014
Muscle	$4.14x10^{-12}$ 0.037	$6.36x10^{-12}$ 0.011	$1.24x10^{-12}$ 0.015	$1.13x10^{-11}$ 0.011	$1.08x10^{-11}$ 0.009
Brain	$5.63x10^{-12}$ 0.133	$9.64x10^{-12}$ 0.047	$1.11x10^{-11}$ 0.024	$6.02x10^{-12}$ 0.089	$1.22x10^{-11}$ 0.040
GI Tract	$1.66x10^{-12}$ 0.244	$2.49x10^{-12}$ 0.073	$1.16x10^{-14}$ 0.612	$6.49x10^{-12}$ 0.073	$6.67x10^{-12}$ 0.064
CPU (min)	10.55	26.89	21.69	20.03	23.83

Note: the numbers below normalized dose are fractional standard deviation.

In all four cases, the normalized doses fluctuate between higher and lower values with respect to the test case (Table VI). All of the values (except gastro-intestinal tract from above, described below) are within acceptable statistical limits of the test case values showing the random nature of a Monte Carlo code. Although the standard deviations are not significantly lower compared with the test case, the CPU time is dramatically reduced in series 2 and 3 (up to 85% reduction). The very slight reduction in variance resulting from the additional neutrons analyzed in the fourth series does not justify the extra CPU time required to run the series. The solution for the gastro-intestinal tract from neutron started above is suspect in all four series and the test case. Neutrons started from above the phantom have many mean-free paths to travel before reaching the gastro-intestinal tract. Since a large majority of the neutrons either scatter out of the phantom or are statistically removed before reaching the tract, poor statistics are the result for this organ. To overcome this, many more neutrons need to be started. Unfortunately the computer system used does not have the buffer space

needed to make such an accommodation.

Sample Calculation.

A sample applications problem demonstrates the use of MORSE phantom results. Consider a 10 kton tactical weapon detonation at sea level on solid, flat ground. The problem is: "what total body dose does a soldier 1.5 km from ground zero receive if the blast is to the soldier's back?" The assumptions required are:

1. the weapon is a plutonium fission device,
2. neutrons do not scatter from the surface of the earth,
3. the atmosphere is homogeneous,
4. the neutrons are mono-directional when they reach the soldier,
5. the neutrons reaching the soldier have an energy distribution as described in Table V,
6. neutrons have an average quality factor per energy group of 10 (12:373).

A $4\pi r^2$ normalized fluence per source neutron of 0.02 (3: Ch.9,18) results from a 1.5 km target range at sea level. The number of neutrons escaping the device is 1.1 X 10^{23} neutrons/kton (3:Ch 9,2). The resulting neutron fluence incident on the soldier is 9.2 X 10^{10} n/cm^2. Figure 6 is a graphical representation of Table VI, the original analysis results. Reading normalized absorbed dose values in the -Y direction for each organ off the graph and multiplying by the incident fluence results in an absorbed dose of approximately 5 Gray-tissue (500 Rad-tissue). To determine an equivalent dose, the absorbed dose is multiplied by the quality factor of 10. The equivalent dose is 50 Sievert (5000 REM), which would render the average human incapacitated within approximately 30 minutes (7:14).

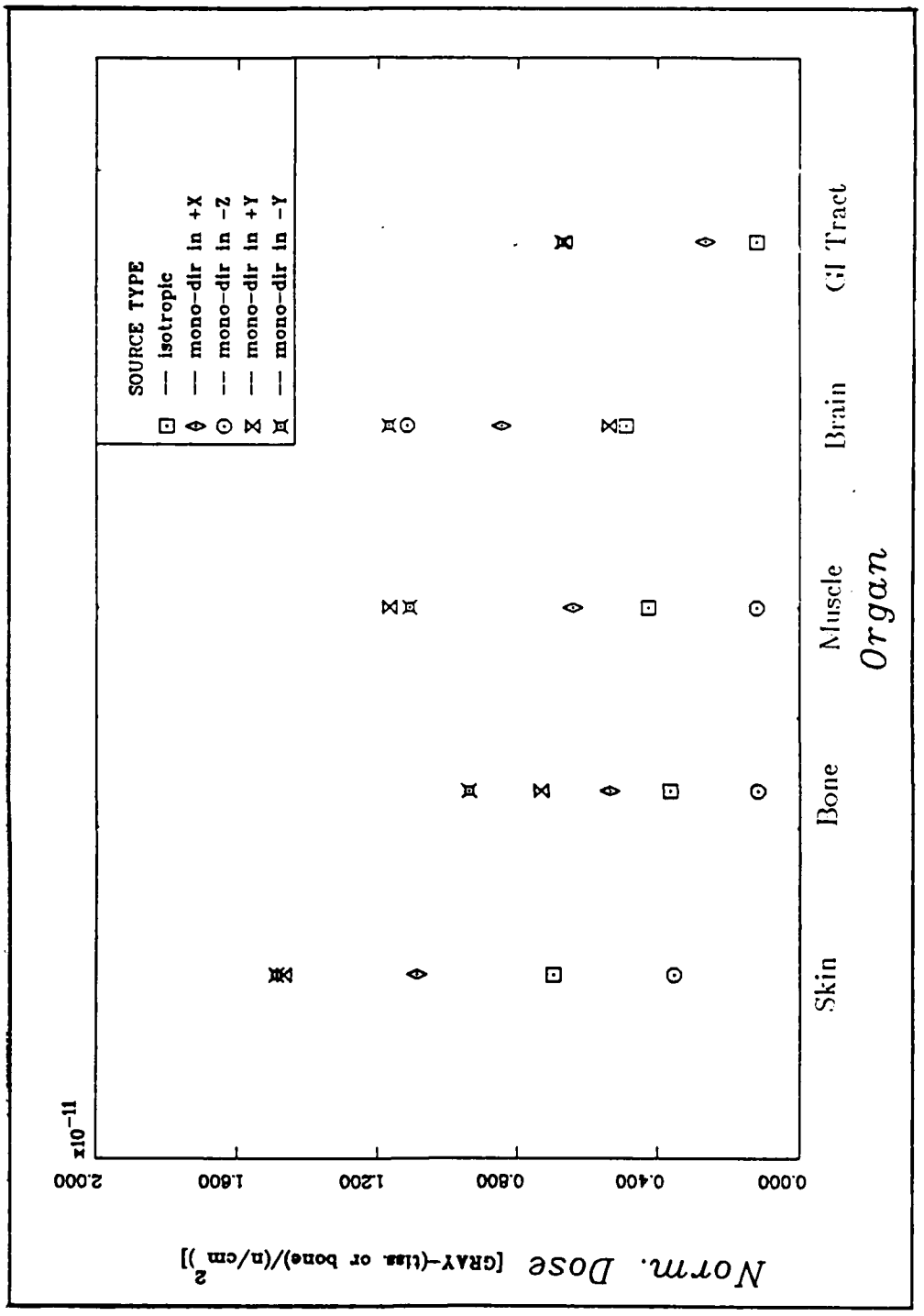

Figure 6. Normalized Absorbed Dose for Test Case

IV. Conclusions and Recommendations

Conclusions

The goal of this study is to provide a fast, convenient, and generic method of estimating absorbed neutron radiation dose to human organs. Modification of the ORNL MORSE code proves to be a useful technique for achieving this goal. Conclusions about the user-ready phantom version of MORSE are as follows:

1) The most efficient MORSE analysis, to date, is use of a 10 microsecond time kill. Time kill reduces analysis time by up to 85% without degrading absorbed dose results or sample variance.

2) Through simple changes to the input data file, normalized absorbed neutron dose estimates are easily achievable for a wide variety of neutron environments. Each analysis requires under 15 CPU minutes.

3) Although this version of MORSE is functional, additional modifications are foreseeable to produce a more efficient and meaningful code. The next section contains recommendations for further modifications.

Recommendations

The following six recommendations could produce a much more efficient and generic phantom code.

- Determine a method of increasing the number of allowable neutrons per batch. This could improve the variance in the smaller organ estimates.
- Follow a systematic approach to reduce the sample variance using splitting, Russian roulette, and path length stretching. Path length stretching is not attempted in this project, because of shortage of time, but may benefit future modifications.
- Implement 37 group (or less) quality factors to produce effective dose in

Sievert.

- Include additional phantom geometries such as kneeling and sitting positions. The additional geometries more closely resemble common battle situations (i.e. tank driver or pilot).
- Include more organs of interest if somatic radiation effects are of concern.
- Include dose from internally produced gamma-rays to more closely estimate actual dose.

Appendix A: Example of MORSE Input Data File

Appendix A is a listing of the input data file (excluding cross sections) required by MORSE. The format of the data file is given in the MORSE Applications Guide (5). The file is set up to produce a fission neutron energy spectrum incident on a three-dimensional phantom.

```
MORSE, PHANTOM GEOMETRY, 37 n groups  (fis. energy spect.) - TEST FILE
 1200 1200    20     1   37     0   37   37    0    0  80.     3     0
    0   37     0    01.0         0.00001   0.          0.0         4380.
       0.0          0.0          0.0          0.0          1.0          1.0          1.0
 0.0000 +0 0.0000 +0 0.0000 +0 0.0000 +0 0.0000 +0 0.0000 +0 0.0000 +0
 0.0000 +0 3.8400 -3 3.5000 -3 5.3900 -3 7.3500 -3 1.8400 -2 3.2500 -3
 8.4700 -3 5.5000 -2 3.2400 -2 1.0600 -2 9.7200 -2 1.4700 -1 2.1600 -1
 1.5000 -1 1.9300 -2 1.2100 -1 5.7300 -2 6.0000 -3 2.4000 -2 1.4400 -2
 0.0000 +0 0.0000 +0 0.0000 +0 0.0000 +0 0.0000 +0 0.0000 +0 0.0000 +0
 0.0000 +0 0.0000 +0
 1.9600 +7 1.6900 +7 1.4900 +7 1.4200 +7 1.3800 +7 1.2800 +7 1.2200 +7
 1.1100 +7 1.0000 +7 9.0500 +6 8.1900 +6 7.4100 +6 6.3800 +6 4.9700 +6
 4.7200 +6 4.0700 +6 3.0100 +6 2.3900 +6 2.3100 +6 1.8300 +6 1.1100 +6
 5.5500 +5 1.5800 +5 1.1100 +5 5.2500 +4 2.4800 +4 2.1900 +4 1.0300 +4
 3.3500 +3 1.2300 +3 5.8300 +2 1.0100 +2 2.9000 +1 1.0700 +1 3.0600 +0
 1.1300 +0 4.1400 -1
000035FA731A
    0    0    0    0    0    6    1
    0    0    0    0
    0    0          Geometry of BODY (skin, bones, AND MUSCLE)
  SPH    1        0.0        2.5       86.5        9.0
  SPH    2        0.0        2.5       86.5        8.8
  RCC    3        0.0       -9.0       80.0        0.0         9.0       -1.0
                  7.5
  RCC    4        0.0       -8.8       80.0        0.0         8.8       -1.0
                  7.3
  RCC    5        0.0        0.0       69.0        0.0         1.0        12.
                  5.2
  RCC    6        0.0        0.0        69.        0.0         1.0        12.
                  5.0
  REC    7        0.0        0.0        0.0        0.0         0.0        70.0
                  0.0       10.5        0.0       15.5         0.0         0.0
  REC    8        0.0        0.0        0.2        0.0         0.0        69.6
                  0.0       10.3        0.0       15.3         0.0         0.0
  TRC    9      -25.0        0.0       23.0       12.0         0.0        45.0
                  3.0        4.0
  TRC   10      -25.0        0.0       23.0       12.0         0.0        45.0
                  2.8        3.8
  TRC   11       25.0        0.0       23.0      -12.0         0.0        45.0
                  3.0        4.0
  TRC   12       25.0        0.0       23.0      -12.0         0.0        45.0
                  2.8        3.8
  TRC   13      -25.0        0.0      -17.0        0.0         0.0        40.0
                  2.0        3.0
  TRC   14      -25.0        0.0      -16.8        0.0         0.0        39.8
                  1.8        2.8
  TRC   15       25.0        0.0      -17.0        0.0         0.0        40.0
                  2.0        3.0
  TRC   16       25.0        0.0      -16.8        0.0         0.0        39.8
                  1.8        2.8
  TRC   17      -10.0        0.0      -40.0        2.5         0.0        42.0
                  6.0        8.5
  TRC   18      -10.0        0.0      -40.0        2.5         0.0        41.8
                  5.8        8.3
  TRC   19       10.0        0.0      -40.0       -2.5         0.0        42.0
                  6.0        8.5
  TRC   20       10.0        0.0      -40.0       -2.5         0.0        41.8
                  5.8        8.3
  TRC   21      -10.0        0.0      -80.0        0.0         0.0        41.0
                  3.5        6.0
```

TRC	22	-10.0	0.0	-79.8	0.0	0.0	40.8
		3.3	5.8				
TRC	23	10.0	0.0	-80.0	0.0	0.0	41.0
		3.5	6.0				
TRC	24	10.0	0.0	-79.8	0.0	0.0	40.8
		3.3	5.8				
SPH	25	0.0	-10.0	0.0	2.5		
SPH	26	0.0	-10.0	0.0	2.3		
SPH	27	0.0	2.5	86.5	8.799		
SPH	28	0.0	2.5	86.5	8.3		
RCC	29	0.0	-8.8	80.0	0.0	8.8	-1.0
		7.3					
RCC	30	0.0	-7.8	80.0	0.0	7.8	-1.0
		6.8					
RCC	31	0.0	2.0	80.0	0.0	5.8	-41.0
		2.0					
RCC	32	0.0	7.8	40.0	0.0	0.0	-20.0
		2.0					
REC	33	0.0	0.0	0.2	0.0	0.0	19.8
		0.0	9.8	0.0	14.8	0.0	0.0
REC	34	0.0	0.0	0.2	0.0	0.0	19.8
		0.0	8.8	0.0	13.8	0.0	0.0
RPP	35	-16.0	16.0	-11.0	-4.0	0.0	20.2
REC	36	0.0	0.0	25.0	0.0	0.0	40.0
		0.0	9.5	0.0	14.5	0.0	0.0
REC	37	0.0	0.0	25.0	0.0	0.0	40.0
		0.0	9.0	0.0	14.0	0.0	0.0
RPP	38	-15.5	15.0	-11.0	11.0	26.7	28.5
RPP	39	-15.5	15.0	-11.0	11.0	30.2	32.0
RPP	40	-15.5	15.0	-11.0	11.0	33.7	35.5
RPP	41	-15.5	15.0	-11.0	11.0	37.2	38.9
RPP	42	-15.5	15.0	-11.0	11.0	40.6	42.4
RPP	43	-15.5	15.0	-11.0	11.0	44.1	45.9
RPP	44	-15.5	15.0	-11.0	11.0	47.6	49.4
RPP	45	-15.5	15.0	-11.0	11.0	51.1	52.9
RPP	46	-15.5	15.0	-11.0	11.0	54.6	56.3
RPP	47	-15.5	15.0	-11.0	11.0	58.0	59.8
RPP	48	-15.5	15.0	-11.0	11.0	61.5	63.3
WED	49	0.0	-15.0	25.0	-7.0	0.0	0.0
		0.0	0.0	20.0	0.0	7.0	0.0
WED	50	0.0	-15.0	25.0	7.0	0.0	0.0
		0.0	0.0	20.0	0.0	7.0	0.0
RPP	51	-1.5	1.5	-14.5	-14.0	45.0	65.0
RCC	52	-10.0	0.0	-41.0	-3.8	0.0	41.0
		2.5					
RCC	53	10.0	0.0	-41.0	3.8	0.0	41.0
		2.5					
RCC	54	-10.0	0.0	-40.0	0.0	0.0	-40.0
		2.0					
RCC	55	10.0	0.0	-40.0	0.0	0.0	-40.0
		2.0					
RCC	56	-14.0	0.0	63.5	-11.0	0.0	-40.5
		1.5					
RCC	57	14.0	0.0	63.5	11.0	0.0	-40.5
		1.5					
RCC	58	-25.0	0.0	23.0	0.0	0.0	-39.8
		1.3					
RCC	59	25.0	0.0	23.0	0.0	0.0	-39.8
		1.3					
RPP	60	-8.5	8.5	-6.0	2.5	78.0	89.5

```
REC  61        0.0       0.0       9.0       0.0       0.0      15.0
               0.0       5.0       0.0       9.0       0.0       0.0
SPH  62        0.0       0.0      10.0     101.0
RPP  63     -120.0     120.0    -120.0     120.0    -120.0     120.0
END
SKN      OR   +1       -4        -6       -2OR   +3       -2        -6       -4OR   +5
              -2       -4        -8       -6OR   +7       -6       -10       -12     -18
             -20       -26       -8OR     +9       -8       -10OR    +11       -8     -12
         OR  +13       -14OR    +15       -16OR  +17       -8       -26       -18OR  +19
              -8       -26       -20OR    +21       -22OR  +23       -24OR  +25       -8
             -17       -19       -25
BON      OR  +27       -30       -28OR    +29       -28      -30OR    +31       -28OR  +32
         OR  +33       -34       -35OR    +36       -37      -38      -39       -40    -41
             -42       -43       -44      -45       -46      -47      -48       -49    -50
         OR  +51OR     +52OR     +53OR    +54OR     +55OR    +56OR    +57OR     +58OR  +59
MUS      OR   +2       -27       -29      -31OR      +4      -29OR     -8       -31    -32
             -33       -36       -51      -61OR     +33      +35       -34OR    +36    +38
             -37OR     +36       +39      -37OR     +36      +40       -37OR    +36    +41
             -37OR     +36       +42      -37OR     +36      +43       -37      -51OR  +36
             +44       -37       -51OR    +36       +45      -37       -51OR    +36    +46
             -37       -51OR     +36      +47       -37      -51OR     +36      +48    -37
             -51OR     +36       +49      -37OR     +36      +50       -37OR    +28    +60
         OR  +30OR     +34       -61OR    +37       -31      -32       -51OR     +6    -31
             -29       -27OR     +26       -8       -18      -20OR     +10       -8    -56
         OR  +12        -8       -57OR    +14       -58OR    +16       -59OR    +18     -8
             -26       -52OR     +20       -8       -26      -53OR     +22      -54    -52
         OR  +24       -55       -53
BRN          +28       -60
GIT          +61
AIR          +62        -1        -3        -5        -7        -9       -11       -13    -15
             -17       -19       -21       -23       -25
ENV          +63       -62
END
  1    2    3    4    5    6    6
  1    2    3    4    5    6    0
PHANTOM PROBLEM 37n, for 11 elements , RUN 3
 37   37    0    0   58   31    4    3    3    3    4    2    0    3
  0    0    0    0    0    0    0  -10    0    0    0
SAMBO ANALYSIS INPUT DATA: phantom problem
  5    1    1    0    0    1    1    1
   0.0       0.0       0.0
   0.0       0.0       0.0
   0.0       0.0       0.0
   0.0       0.0       0.0
   0.0       0.0       0.0
NORMALIZED ORGAN DOSES
GRAY-tissue/(s-n/cm**2) [DET:1,3,4,5] or GRAY-BONE/(s-n/cm**2) [DET:2]
      1.0       1.0       1.0       1.0       1.0       1.0       1.0
      1.0       1.0       1.0       1.0       1.0       1.0       1.0
      1.0       1.0       1.0       1.0       1.0       1.0       1.0
      1.0       1.0       1.0       1.0       1.0       1.0       1.0
      1.0       1.0       1.0       1.0       1.0       1.0       1.0
      1.0       1.0
USER DEFINED INPUT: SOURCE TYPE AND X-SECTION MEDIA
 1
  1  2  1  1  1  3
   3340.   6710.  49570.   1555.   2120.  3.53E6
$$$$$$$$$$ PHANTOM DOSE FROM INPUT SPECTRUM **********
```

Appendix B: Final Version of Modified MORSE Subroutines

Appendix B contains the MORSE driver program and six subroutines modified to analyze the human phantom model. The subroutines were either modified or completely re-written to provide a phantom specific code. The subroutines are:

GTMED
- determines type of material within a zone (bone, tissue, air)

BANKR
- bookkeeping subroutine

TRKBDR
- determines dose contributions from particles crossing a boundary

TRKCOL
- determines dose contributions from particles undergoing a collision

INSCOR
- reads additional input data specific to this study

SOURCE
- produces a neutron source incident on the phantom either isotropically from a sphere surrounding the phantom, or mono-directionally from the right side, above, front, or back of the phantom.

Detailed descriptions of all MORSE subroutines are in the Applications Guide (5). Instructions on the use of MORSE are in the thesis notebook.

```
C       PHANTOM.FOR
C*************************************************************************
C*  Modifications by:                                                    *
C*       Capt Michael G. Archuleta                                       *
C*       GNE-88M                                                         *
C*           modifications done Oct/Nov 1987 for thesis project          *
C*           to determine organ dose as a function of n flux.            *
C*         This version does not determine uncollided fluence.           *
C*         Instead, a tracklength estimater is used to determine         *
C*         fluence and is called for collisions and boundary crossings.  *
C*         Fluence is subsequently converted to normalized dose in       *
C*         GRAY/(n/cm**2) of tissue or bone.                             *
C*                                                                       *
C*         This version requires the PHANTOM.DAT input data file,        *
C*         PHANTSRC.FOR source generator, and MED3.BIN binary cross-     *
C*         section data file.                                            *
C*************************************************************************
C * * THIS IS THE MAIN ROUTINE * * * * * * * * * * * * * * * * * * * * *
C * *
C * * THE FOLLOWING CARD DETERMINES THE SIZE ALLOWED FOR BLANK COMMON  *
      COMMON NC(50000)
C * *  (REGION SIZE NEEDED IS ABOUT 150K + 4*(SIZE OF BLANK COMMON IN WO
C * * NOTE - THE ORDER OF COMMONS IN THIS ROUTINE IS IMPORTANT AND MUST
C * *   POND TO THE ORDER USED IN DUMP ROUTINES SUCH AS HELP, XSCHLP, AN
C * *
C * * LABELLED COMMONS FOR WALK ROUTINES    * * * * * * * * * * * * * * *
      COMMON /APOLLO/ AGSTRT,DDF,DEADWT(26),ITCUT,ITIN
      COMMON /FISBNK/ MFISTP
      COMMON /NUTRON/ NAME
C * *
C * * LABELLED COMMONS FOR CROSS-SECTION ROUTINES  * * * * * * * * * * *
      COMMON /LOCSIG/ ISCCOG
      COMMON /MEANS/ NM
      COMMON /MOMENT/ NMOM
      COMMON /QAL/ Q
      COMMON /RESULT/ POINT
C * *
C * * LABELLED COMMONS FOR GEOMETRY INTERFACE ROUTINES  * * * * * * * *
      COMMON /GEOMC/ XTWO
      COMMON /NORMAL/ UNORM
C * *
C * * LABELLED COMMONS FOR USER ROUTINES  * * * * * * * * * * * * * * *
      COMMON /PDET/ ND
      COMMON /USER/ AGST
C * *
C * * COMMON /DUMMY/ WILL NOT BE FOUND ELSEWHERE IN THE PROGRAM  * * * *
      COMMON /DUMMY/ DUM
C * *
C * * LABELLED COMMONS FOR USER INPUT DATA * * * * * * * * *
      COMMON /INSDAT/ISRC,VOLI(10),IXSECM(10)
        CHARACTER*20, NAM1
        CHARACTER*20, NAM2
        TYPE *,' '
        TYPE *,'********* MORSE Code, HUMAN PHANTOM Problem *********'
        TYPE *,'          --------> WARNING !!! <--------'
        TYPE *,'ABORT if mixed x-secs are not assigned to FOR010'
        TYPE *,' '
        TYPE *,'ENTER NAME OF INPUT FILE'
        ACCEPT 100, NAM1
100     FORMAT(A20)
```

```
        TYPE *,'ENTER NAME OF OUTPUT FILE'
        ACCEPT 200, NAM2
200     FORMAT (A20)
        OPEN(UNIT=1,NAME=NAM1,TYPE='OLD')
        OPEN(UNIT=2,NAME=NAM2,TYPE='NEW')
      ITOUT = 2
      ITIN = 1
        NLFT=34999
      CALL MORSE(NLFT)
        TYPE 300, NAM2
300     FORMAT(X,'OUTPUT FILE IS ',A20)
      STOP
      END
C********************* subroutine GTMED*******************
      SUBROUTINE GTMED(MDGEOM,MDXSEC)
C * * * * * * * * *
C ** This version of GTMED assigns xsection media to geometry media
C ** as specified in the additional user input lines in the input
C ** data file
C * * * * * * * * *
      COMMON /INSDAT/ISRC,VOLI(10),IXSECM(10)
      IF(MDGEOM.EQ.0.OR.MDGEOM.EQ.1000)GO TO 90
      MDXSEC = IXSECM(MDGEOM)
      RETURN
   90 MDXSEC = MDGEOM
      RETURN
      END
C **************FUNCTION DIREC******************
      FUNCTION DIREC(F)
C * * * * * * * *
C ** This version of DIREC is a dummy routine that does
C ** absolutely nothing.
C * * * * * * * *
      direc=1.
      RETURN
      END
C***************SUBROUTINE BANKR*****************
      SUBROUTINE BANKR(NBNKID)
C * * * * * * * * * *
C ** This version call TRKBDR from BANKR(7) and TRKCOL from BANKR(5).
C * * * * * * * * * *
      COMMON /APOLLO/ AGSTRT,DDF,DEADWT(5),ETA,ETATH,ETAUSD,UINP,VINP,
     1 WINP,WTSTRT,XSTRT,YSTRT,ZSTRT,TCUT,XTRA(10),
     2   I0,I1,MEDIA,IADJM,ISBIAS,ISOUR,ITERS,ITIME,ITSTR,LOCWTS,LOCFWL,
     3 LOCEPR,LOCNSC,LOCFSN,MAXGP,MAXTIM,MEDALB,MGPREG,MXREG,NALB,
     4 NDEAD(5),NEWNM,NGEOM,NGPQT1,NGPQT2,NGPQT3,NGPQTG,NGPQTN,NITS,
     5 NKCALC,NKILL,NLAST,NMEM,NMGP,NMOST,NMTG,NOLEAK,NORMF,NPAST,
     6 NPSCL(13),NQUIT,NSIGL,NSOUR,NSPLT,NSTRT,NXTRA(10)
      COMMON /NUTRON/ NAME,NAMEX,IG,IGO,NMED,MEDOLD,NREG,U,V,W,UOLD,VOLD
     1 ,WOLD,X,Y,Z,XOLD,YOLD,ZOLD,WATE,OLDWT,WTBC,BLZNT,BLZON,AGE,OLDAGE
      NBNK = NBNKID
      IF (NBNK) 100,100,140
  100 NBNK = NBNK + 5
      GO TO (104,103,102,101),NBNK
  101 CALL STRUN
C     CALL HELP(4HSTRU,1,1,1,1)
      RETURN
  102 NBAT = NITS - ITERS
      NSAVE = NMEM
      CALL STBTCH(NBAT)
```

```
C   NBAT IS THE BATCH NO. LESS ONE
      RETURN
  103 CALL NBATCH(NSAVE)
C   NSAVE IS THE NO. OF PARTICLES STARTED IN THE LAST BATCH
      RETURN
  104 CALL NRUN(NITS,NQUIT)
C   NITS IS THE NO. OF BATCHES COMPLETED IN THE RUN JUST COMPLETED
C   NQUIT .GT. 1 IF MORE RUNS REMAIN
C         .EQ. 1 IF THE LAST SCHEDULED RUN HAS BEEN COMPLETED
C         IS THE NEGATIVE OF THE NO. OF COMPLETE RUNS, WHEN AN
C              EXECUTION TIME KILL OCCURS
      RETURN
  140 GO TO (1,2,3,4,5,6,7,8,9,10,11,12,13),NBNK
C  NBNKID   COLL TYPE   BANKR CALL      NBNKID   COLL TYPE   BANKR CALL
C     1       SOURCE     YES (MSOUR)       2       SPLIT      NO  (TESTW)
C     3       FISSION    YES (FPROB)       4       GAMGEN     YES (GSTORE
C     5       REAL COLL  YES (MCRSE)       6       ALBEDO     YES (MCRSE)
C     7       BDRYX      YES (NXTCOL)      8       ESCAPE     YES (NXTCOL
C     9       E-CUT      NO  (MCRSE)      10       TIME KILL  NO  (MCRSE)
C    11       R R KILL   NO  (TESTW)      12       R R SURV   NO  (TESTW)
C    13       GAMLOST    NO  (GSTORE)
    1 RETURN
    2 RETURN
    3 RETURN
    4 RETURN
    5 CALL TRKCOL
      RETURN
    6 RETURN
    7 CALL TRKBDR
      RETURN
    8 RETURN
    9 RETURN
   10 RETURN
   11 RETURN
   12 RETURN
   13 RETURN
      END
C************************ SUBROUTINE TRKBDR ********************
      SUBROUTINE TRKBDR
C***********************************************************************
C   this version determines flux as tracklength divided by detector
C   volume for all boundary crossing out of body organs.
C         the variables used are:
C               AREA   - surface area of neutron source region [cm**2]
C               CON    - dose contribution of n in MEDOLD [GRAY/(n/cm**2)]
C               ISRC   - type of source defined in input data file
C               KERMAB - bone KERMA factor [RAD-bone/(n/cm**2)]
C               KERMAT - tissue KERMA factor [RAD-tiss/(n/cm**2)]
C               MEDOLD - media (organ) n is exitting
C               MXREG  - highest region number (corresponds to air)
C               TRK    - tracklength from previous collision/boundary
C                        crossing to current boundary crossing [cm]
C               VOL    - volume of media (organ) n is exitting
C***********************************************************************
      COMMON /APOLLO/ AGSTRT,DDF,DEADWT(5),ETA,ETATH,ETAUSD,UINP,VINP,
     1 WINP,WTSTRT,XSTRT,YSTRT,ZSTRT,TCUT,XTRA(10),
     2   I0,I1,MEDIA,IADJM,ISBIAS,ISOUR,ITERS,ITIME,ITSTR,LOCWTS,LOCFWL,
     3 LOCEPR,LOCNSC,LOCFSN,MAXGP,MAXTIM,MEDALB,MGPREG,MXREG,NALB,
     4 NDEAD(5),NEWNM,NGEOM,NGPQT1,NGPQT2,NGPQT3,NGPQTG,NGPQTN,NITS,
     5 NKCALC,NKILL,NLAST,NMEM,NMGP,NMCST,NMTG,NOLEAK,NORMF,NPAST,
```

```
     6 NPSCL(13),NQUIT,NSIGL,NSOUR,NSPLT,NSTRT,NXTRA(10)
      COMMON /NUTRON/ NAME,NAMEX,IG,IGO,NMED,MEDOLD,NREG,U,V,W,UOLD,VOLD
     1 ,WOLD,X,Y,Z,XOLD,YOLD,ZOLD,WATE,OLDWT,WTBC,BLZNT,BLZON,AGE,OLDAGE
      COMMON /INSDAT/ISRC,VOLI(10),IXSECM(10)
      REAL KERMAT(37),KERMAB(37),AREA(5)
      DATA AREA/62800.,4580.,1560.,10600.,10600./
      DATA KERMAT/.7225E-08,.6858E-08,.6642E-08,.6528E-08,.6380E-08,
     *.6188E-08,.6118E-08,.5857E-08,.5694E-08,.5393E-08,.5284E-08,
     *.5039E-08,.4723E-08,.4479E-08,.4412E-08,.4193E-08,.3560E-08,
     *.3264E-08,.3200E-08,.2761E-08,.2097E-08,.1302E-08,.8010E-09,
     *.5585E-09,.3148E-09,.2145E-09,.1472E-09,.6269E-10,.2211E-10,
     *.9187E-11,.3104E-11,.1027E-11,.9187E-12,.1341E-11,.2051E-11,
     *.3689E-11,.1515E-10/
      DATA KERMAB/.5511E-08,.5196E-08,.5016E-08,.4929E-08,.4804E-08,
     *.4680E-08,.4608E-08,.4396E-08,.4272E-08,.3991E-08,.3911E-08,
     *.3663E-08,.3405E-08,.3199E-08,.3162E-08,.2995E-08,.2489E-08,
     *.2256E-08,.2214E-08,.1906E-08,.1445E-08,.8942E-09,.5470E-09,
     *.3808E-09,.2144E-09,.1461E-09,.1002E-09,.4276E-10,.1521E-10,
     *.6449E-11,.2407E-11,.1196E-11,.1415E-11,.2223E-11,.3746E-11,
     *.6100E-11,.2474E-10/
C * * * check for neutron coming from VOID, ENV, or AIR * * *
      IF(MEDOLD.EQ.MXREG.OR.MEDOLD.EQ.0)RETURN
C ***** add track length to zone detector * * ****************
C   FLUENCE [CON] =TRK/VOL                                    *
C   Norm.DOSE [rad/(n/cm**2)] = fluence * kerma factor*area   *
C   Gray/(n/cm**2) = fluence * kerma factor *area/100.        *
C ************************************************************
      TRK = WATE * SQRT((X-XOLD)**2 + (Y-YOLD)**2 + (Z-ZOLD)**2)
      VOL = VOLI(MEDOLD)
      IF(MEDOLD.EQ.2)CON = TRK * KERMAB(IG) * AREA(ISRC)/(VOL*100.)
      IF(MEDOLD.NE.2)CON = TRK * KERMAT(IG) * AREA(ISRC)/(VOL*100.)
C  if source is isotropic, correct for 2pi production
      IF(ISRC.EQ.1)CON = CON/2.
      CALL FLUXST(MEDOLD,IG,CON,0.0,0.0,0)
C  *  *  SWITCH =  0  -- STORE IN ALL RELEVANT ARRAYS EXCEPT UD
      RETURN
      END
C********************* SUBROUTINE TRKCOL ******************
      SUBROUTINE TRKCOL
C***********************************************************************
C   this version determines flux, at each collision site,
C   from tracklength divided by detector
C   volume and in used with TRKBDR (called from BANKR(7))
C       Variables used are identical to TRKBDR variables except:
C             NMED  = current media (organ) n is in.
C             IGO   = n energy before collision
C***********************************************************************
      COMMON /APOLLO/ AGSTRT,DDF,DEADWT(5),ETA,ETATH,ETAUSD,UINP,VINP,
     1 WINP,WTSTRT,XSTRT,YSTRT,ZSTRT,TCUT,XTRA(10),
     2   I0,I1,MEDIA,IADJM,ISBIAS,ISOUR,ITERS,ITIME,ITSTR,LOCWTS,LOCFWL,
     3 LOCEPR,LOCNSC,LOCFSN,MAXGP,MAXTIM,MEDALB,MGPREG,MXREG,NALB,
     4 NDEAD(5),NEWNM,NGEOM,NGPQT1,NGPQT2,NGPQT3,NGPQTG,NGPQTN,NITS,
     5 NKCALC,NKILL,NLAST,NMEM,NMGP,NMOST,NMTG,NOLEAK,NORMF,NPAST,
     6 NPSCL(13),NQUIT,NSIGL,NSOUR,NSPLT,NSTRT,NXTRA(10)
      COMMON /NUTRON/ NAME,NAMEX,IG,IGO,NMED,MEDOLD,NREG,U,V,W,UOLD,VOLD
     1 ,WOLD,X,Y,Z,XOLD,YOLD,ZOLD,WATE,OLDWT,WTBC,BLZNT,BLZON,AGE,OLDAGE
      COMMON /INSDAT/ISRC,VOLI(10),IXSECM(10)
      REAL KERMAT(37),KERMAB(37),AREA(5)
      DATA AREA/62800.,4580.,1560.,10600.,10600./
      DATA KERMAT/.7225E-08,.6858E-08,.6642E-08,.6528E-08,.6380E-08,
```

```
     *.6188E-08,.6118E-08,.5857E-08,.5694E-08,.5393E-08,.5284E-08,
     *.5039E-08,.4723E-08,.4479E-08,.4410E-08,.4193E-08,.3560E-08,
     *.3264E-08,.3200E-08,.2761E-08,.2097E-08,.1302E-08,.8010E-09,
     *.5585E-09,.3148E-09,.2145E-09,.1472E-09,.6269E-10,.2211E-10,
     *.9187E-11,.3104E-11,.1027E-11,.9187E-12,.1341E-11,.2251E-11,
     *.3689E-11,.1515E-10/
      DATA KERMAB/.5511E-08,.5196E-08,.5016E-08,.4929E-08,.4824E-08,
     *.4680E-08,.4608E-08,.4396E-08,.4272E-08,.3991E-08,.3911E-08,
     *.3663E-08,.3405E-08,.3199E-08,.3162E-08,.2995E-08,.2489E-08,
     *.2256E-08,.2214E-08,.1906E-08,.1445E-08,.8942E-09,.5470E-09,
     *.3808E-09,.2144E-09,.1461E-09,.1002E-09,.4276E-10,.1521E-10,
     *.6449E-11,.2407E-11,.1196E-11,.1415E-11,.2223E-11,.3746E-11,
     *.6100E-11,.2474E-10/
C * * * check for ENV OR AIR
      IF(NMED.EQ.MXREG.OR.NMED.EQ.0)RETURN
C * * * calculate fluence estimate
      TRK = WTBC * SQRT((X-XOLD)**2 + (Y-YOLD)**2 + (Z-ZOLD)**2)
      VOL = VOLI(NMED)
      IF(NMED.EQ.2)CON = TRK * KERMAB(IGO) * AREA(ISRC)/(VOL*100.)
      IF(NMED.NE.2)CON = TRK * KERMAT(IGO) * AREA(ISRC)/(VOL*100.)
      IF(ISRC.EQ.1)CON = CON/2.0
      CALL FLUXST(NMED,IGO,CON,0.0,0.0,0)
C  *   *  SWITCH =  0  -- STORE IN ALL RELEVANT ARRAYS EXCEPT UD
      RETURN
      END
*****************SUBROUTINE INSCOR***************
      SUBROUTINE INSCOR
C***********************************************************
C This version reads in 4 lines of user input data from the input
C data file.
C    LINE 1 - dummy comment line
C    LINE 2 - The type of source incident on the phantom:
C             1 - isotropic
C             2 - source plane in +X direction
C             3 - source plane in -X direction
C             4 - source plane in +Y direction
C             5 - source plane in -Y direction
C    LINE 3 - assigns cross-section media to geometry media
C    LINE 4 - defines the volume of each region (organ)
C***********************************************************
      COMMON /APOLLO/ AGSTRT,DDF,DEADWT(5),ETA,ETATH,ETAUSD,UINP,VINP,
     1 WINP,WTSTRT,XSTRT,YSTRT,ZSTRT,TCUT,XTRA(10),
     2   I0,I1,MEDIA,IADJM,ISBIAS,ISOUR,ITERS,ITIME,ITSTR,LOCWTS,LOCFWL,
     3 LOCEPR,LOCNSC,LOCFSN,MAXGP,MAXTIM,MEDALB,MGPREG,MXREG,NALB,
     4 NDEAD(5),NEWNM,NGEOM,NGPQT1,NGPQT2,NGPQT3,NGPQTG,NGPQTN,NITS,
     5 NKCALC,NKILL,NLAST,NMEM,NMGP,NMOST,NMTG,NOLEAK,NORMF,NPAST,
     6 NPSCL(13),NQUIT,NSIGL,NSOUR,NSPLT,NSTRT,NXTRA(10)
      COMMON /INSDAT/ISRC,VOLI(10),IXSECM(10)
      CHARACTER*80 NSTRING
      CHARACTER*15 TYPSRC
      READ(I1,10)NSTRING
   10 FORMAT(A80)
      READ(I1,11)ISRC
   11 FORMAT(I2)
      IF(ISRC.EQ.1)TYPSRC = 'ISOTROPIC.'
      IF(ISRC.EQ.2)TYPSRC = 'MONO-DIR IN +X.'
      IF(ISRC.EQ.3)TYPSRC = 'MONO-DIR IN -Z.'
      IF(ISRC.EQ.4)TYPSRC = 'MONO-DIR IN -Y.'
      IF(ISRC.EQ.5)TYPSRC = 'MONO-DIR IN -Y.'
      WRITE(I0,21)TYPSRC
```

```
21 FORMAT(/5X,'**********ADDITIONAL DATA READ THROUGH INSCOR******'//,
  *8X,'The source being used is ',A15)
   READ(I1,12)(IXSECM(I),I=1,MXREG)
12 FORMAT(15I3)
   WRITE(I0,22)(IXSECM(I),I=1,MXREG)
22 FORMAT(/8X,'THE CROSS-SECTION MEDIA OF EACH REGION ARE:'/
  *        12X,15I3)
   READ(I1,13)(VOLI(I),I=1,MXREG)
13 FORMAT(10E8.3)
   WRITE(I0,23)(VOLI(I),I=1,MXREG)
23 FORMAT(/8X,'THE VOLUMES OF EACH REGION ARE:'/
  *       12X,5E8.3/12X,5E8.3)
   RETURN
   END
```

```
      SUBROUTINE SOURCE(IG,U,V,W,X,Y,Z,WATE,MED,AG,ISOUR,ITSTR,NGPQT3,
     1 DDF,ISBIAS,NMTG)
      COMMON /USER/ DUM(9),I0,I1,IDUM(12)
      COMMON WTS(1)
      COMMON /INSDAT/ISRC,VOLI(10),XSEC(10)
C*********************************************************************
C   This version produces a source defined in the user input section
C   of the input data file. The possible source types are:
C         ISRC = 1 - isotropic
C                2 - source plane in +X direction
C                3 - source plane in -Z direction
C                4 - source plane in -Y direction
C                5 - source plane in -Y direction
C*********************************************************************
      DATA ICALL/1/
      IF (ICALL) 10,10,5
 5    ICALL = 0
      WRITE (I0,1000)
 1000 FORMAT (' YOU ARE USING THE DEFAULT VERSION OF SOURCE WHICH SETS W
     1ATE TO DDF AND PROVIDES AN ENERGY IG.')
 10   IF(ISOUR)15,15,60
 15   WATE=DDF
      IF (ISBIAS) 20,20,25
 20   NWT = 2*NMTG
      GO TO 30
 25   NWT = 3*NMTG
 30   R = FLTRNF(0)
      DO 35 I=1,NGPQT3
      IF (R - WTS(I+NWT)) 40,40,35
 35   CONTINUE
 40   IG=I
      IF (ISBIAS) 60,60,45
 45   IF (I-1) 60,50,55
 50   WATE = WATE*WTS(2*NMTG+1)/WTS(3*NMTG+1)
      GO TO 60
 55   WATE = WATE*(WTS(2*NMTG+I)-WTS(2*NMTG+I-1))/(WTS(3*NMTG+I)-WTS(3*N
     1MTG+I-1))
C       *******************************
C       User defined source description
C       *******************************
   60 GO TO (61,62,63,64,65),ISRC
C ********** isotropic source ***********
   61 CALL GTISO(XX,YY,ZZ)
      X    = 100. * XX
      Y    = 100. * YY
      Z    = 100. * ZZ + 10.
   71 CALL GTISO(U,V,W)
      DOTP = X*U + Y*V + Z*W
      IF(DOTP.GE.0.0)GO TO 71
      RETURN
C ********* source plane (+X) ***********
   62 Y    = 26. * FLTRNF(0) - 13.
      Z    = 176.* FLTRNF(0) - 80.
      X    = -30.
      U    = 1.0
      V    = 0.0
      W    = 0.0
      RETURN
C ******** source plane (-Z) ***********
   63 X    = 60. * FLTRNF(0) - 30.
```

```
      Y     = 26. * FLTRNF(0) - 13.
      Z     = 96.
      U     = 0.0
      V     = 0.0
      W     =-1.0
      RETURN
C ********* source plane (+Y) ***********
   64 X     = 60. * FLTRNF(0) - 30.
      Z     = 176.* FLTRNF(0) - 80.
      Y     = -13.
      U     = 0.0
      V     = 1.0
      W     = 0.0
      RETURN
C ********* source plane (-Y) ***********
   65 X     = 60. * FLTRNF(0) - 30.
      Z     = 176.* FLTRNF(0) - 80.
      Y     = 13.
      U     = 0.0
      V     =-1.0
      W     = 0.0
      END
```

BIBLIOGRAPHY

1. Archuleta, Capt Michael G. Thesis notebook compiled in NENG 7.99, Independent Study. School of Engineering, Air Force Institute of Technology (AU), Wright-Patterson AFB OH, July-December 1987.

2. Bartine, D.E. et al. *Production and Testing of the DNA Few-Group Coupled Neutron-Gamma Cross-Section Library.* Contract W-7405-eng-26. Oak Ridge TN: ORNL, March 1977

3. Bridgman, Charles J. Class notes distributed in NENG 6.31, Prompt Effects of Nuclear Weapons. School of Engineering, Air Force Institute of Technology (AU), Wright-Patterson AFB, OH, May 1987.

4. Carter, L.L. and E.D. Cashwell. *Particle-Transport Simulation with Monte-Carlo Method,* Oak Ridge: Technical Information Center, 1977.

5. Cramer, S.N. *Applications Guide to the MORSE Monte Carlo Code.* Contract DE-AC05-840R21400. Oak Ridge TN: ORNL, May 1985.

6. Egbert, Stephen D. et al. "Organ Dosimetry for Japanese A-Bomb Survivors," *Proceedings of the Topical Conference on Theory and Practices in Radiation Protection and Shielding,* Vol. 2, 610-619 Knoxville: American Nuclear Society,1987.

7. Eisenbud, M. *Environmental Radioactivity,* third ed. New York: Academic Press, Inc., 1987.

8. Fisher, H.L. Jr. and W.S. Snyder. "Distribution of Dose in the Body from a Source of Gamma Rays Distributed Uniformly in an Organ," *Radiation Protection,* Part 2, 1473-1486. New York: Pergamon Press Ltd., 1968.

9. Gray, Henry. *Gray's Anatomy* (Twenty-Ninth Edition). Philadelphia: Lea and Febiger, 1973.

10. International Commission of Radiation Protection. *Report of the Task Group on Reference Man.* Publication 23. Pergamon Press Ltd., 1975.

11. Kaul, Dean C. et al. *Phantom Dosimetry Calculations for Use in Radiation Effects Correlations,* Contract DNA 001 83-C-0187. Schaumburg, IL: Science Applications, Inc., 30 July 1984 (AD-A162706).

12. Lamarsh, John R. *Introduction to Nuclear Engineering,* Reading: Addison Wesely, 1977.

13. Radiation Shielding Information Center. *RSIC Computer Code Collection, Morse-CG.* Oak Ridge: ORNL, 24 August 1987.

14. Richmond, Capt John, M.D. Personal interviews. Wright-Patterson AFB Hospital, Wright-Patterson AFB OH, 1 August through 12 December 1987.

15. Schaeffer, N.M. *Reactor Shielding for Nuclear Engineers,* Springfield: National Technical Information Center, 1973.

16. Wood, James. *Computational Methods in Reactor Shielding,* New York: Pergamon Press, 1982.

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Abstract:

This report describes a computer method of determining absorbed neutron dose to a human phantom. Modifications to the Oak Ridge National Laboratory MORSE Monte Carlo code result in a code capable of estimating absorbed dose on a human phantom in the standing position. The phantom organs analyzed are the skin, bone, brain, gastro-intestinal tract, and all remaining tissue. The organ choices are limited to organs capable of incapacitiating a human. The code allows for five different source direction configurations that simulate neutrons, of any specified energy distribution, incident on the phantom.

MORSE analysis of a fission neutron spectrum on the phantom produces absorbed dose estimates comparable with Japanese atomic bomb survivor dose estimates by Scientific Applications International Corporation. The analysis of 24,000 source neutrons requires less than 15 central processing unit minutes on a VAX 11/780 computer (VMS operating system). Although the code is currently usable, additional phantom model orientations, energy-dependent quality factors, and implementation of secondary gamma-ray dose estimation could greatly improve the flexibility and usefulness.

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